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POST-IMPLEMENTATION AUDIT REPORT FOR WISCONSIN PUBLIC SERVICE CORPORATION'S KEWAUNEE NUCLEAR POWER PLANT SAFETY PARAMETER DISPLAY SYSTEM

Technical Evaluation Report

SAIC
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TABLE OF CONTENTS

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FOR
WISCONSIN PUBLIC SERVICE CORPORATION'S
KEWAUNEE NUCLEAR POWER PLANT
SAFETY PARAMETER DISPLAY SYSTEM

1.0 INTRODUCTION

This report documents the findings of the Nuclear Regulatory Commission (NRC) post-implementation audit of the Wisconsin Public Service Corporation's (WPS) Kewaunee Nuclear Power Plant Safety Parameter Display System (SPDS). The audit was conducted October 7-8, 1986. The purpose of the audit was to ascertain that the SPDS met the minimum requirements of Supplement 1 to NUREG-0737 (Reference 1), had been installed in accordance with the licensee's plan, and was functioning properly. The audit team consisted of an NRC team leader, the NRC Kewaunee project manager, another member of the NRC staff, two NRC contractor personnel from Science Applications International Corporation (SAIC), and a representative from SAIC's subcontractor, COMEX Corporation. In addition, the Senior Resident Inspector was present throughout the audit. The team was comprised of individuals representing the disciplines of nuclear systems engineering, nuclear power plant operations, human engineering, and software systems engineering. All members of the team were familiar with the NRC SPDS requirements and the NRC/WPS background documentation. A list of meeting attendees is provided as Attachment 1 to this report.

The findings of the SPDS audit follow a brief review of the background of the SPDS and the regulatory requirements.

2.0 BACKGROUND

All holders of operating licenses issued by the NRC and applicants for an operating license must provide an SPDS in the control room of their plant. The NRC 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 the critical plant variables to the control room operators to aid them in rapidly and reliably determining the safety status of the plant. Supplement 1 to NUREG-0737 requires licensees and applicants to prepare a written safety analysis report (SAR) 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 must 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 (V&V) Plan. The SAR and Implementation Plan are to be submitted to the NRC for staff review. The results of the staff's review are to be published in a Safety Evaluation Report (SER).

By letter dated September 2, 1983 (Reference 2), WPS submitted an SAR regarding the SPDS for the Kewaunee plant. Additional information was provided by letters dated August 1, 1984 (Additional Information on SPDS) (Reference 3); April 26, 1985 (Status of SPDS) (Reference 4); July 14, 1986 (RITS) (Reference 5); and August 21, 1986 (Additional Information on SPDS) (Reference 6).

Incident to the Kewaunee Detailed Control Room Design Review (DCRDR), the NRC staff found the Kewaunee isolation devices to be acceptable and documented the finding in an internal letter dated October 24, 1984. No SER has been issued on the Kewaunee SPDS.

3.0 REGULATORY BASIS FOR SPDS AUDITS

The purpose of the SPDS as stated in NUREG-0737 Supplement 1 establishes the basic functional requirement for the system: "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 operations, the principal purpose and function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. This can be particularly important during anticipated transients and the initial phase of an accident."

The SPDS requirements as defined by Supplement 1 to NUREG-0737 are:

1. To provide a concise display of critical plant variables to control room operators. (para 4.1.a)

2. To be located convenient to control room operators. (para 4.1.b)

3. To continuously display plant safety status information. (para 4.1.b)

4. To be reliable. (para 4.1.b)

5. To be suitably isolated from electrical or electronic interference with safety systems. (para 4.1.c)

6. To be designed incorporating accepted Human Factors Engineering principles. (para 4.1.e)

7. To display, as a minimum, information sufficient to determine plant safety status with respect to five safety functions. (para 4.1.f)

   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

   The five functions listed above will be referred to as critical safety functions (CSFs). Each CSF is depicted by combinations of individual parameters such as steam generator level or cold leg temperature. For audit purposes, the term "variable" will not be used.

8. To implement procedures and operator training addressing actions with and without SPDS. (para 4.1.c)

Guidance as to what constitutes acceptable implementation of the above requirements is provided by Appendix A to NUREG-0800 section 18.2 (Reference 7) and other documents cited therein, particularly NUREG-0700 (Reference 8).

In 1985, an NRC survey of six operating SPDSs was performed to investigate the status and progress of SPDS implementation. The survey included onsite evaluations of licensee documentation and hardware, as well as interviews with operations personnel. The survey findings, including identification of major deficiencies, were distributed in IE Information Notice No. 86-10: Safety Parameter Display System Malfunctions, dated February 13, 1986 (Reference 9).

Since significant SPDS concerns were identified during the 1985 survey of six SPDSs, the NRC concluded that data should be collected at four additional plants in order to determine the need for post-implementation audits. The Kewaunee nuclear power plant was one of the four plants selected for the data collection effort. Prior to the NRC visit, a list of ten SPDS-related questions was sent to the licensee. The licensee's response to the ten questions for the Kewaunee plant was forwarded to NRC by letter dated August 21, 1986 (see Attachment 2).

The audit was designed to evaluate the operational performance of the SPDS as well as its regulatory compliance with Supplement 1 to NUREG-0737. This report reflects the consolidated findings of the audit team.

4.0 REVIEW OF SPDS EVALUATION TOPICS

4.1 Critical Safety Functions/Parameter Selection

In the WPS SAR, as modified by WPS letter of August 1, 1984, the licensee tabulated monitored and displayed parameters, by CSF. The CSF listing was identical with those listed in Supplement 1 to NUREG-0737. The tabulation provided was:

CSF	Monitored Parameter	Displayed Parameter
Reactivity Control	(SR, IR & PR monitor) Power Rx trip status	(SR, IR & PR monitor) Power Rx trip status
Rx Core Cooling & Heat Removal from the primary system	Rx vessel level	Rx vessel level
	Pressurizer level	Pressurizer level
	Core exit temperature	Core exit temperature
	Cold leg temperature	Cold leg temperature
	Hot & cold leg temp.	Rx coolant av.temp.
	Rx coolant loop flows	Rx coolant pump status
	Core exit temp. & Rx coolant pressure	Level of subcooling
	Steam Generator (SG) level	SG level
	SG Pressure	SG Pressure
	Aux feed flow	Aux feed flow
	SG steam flow	SG steam flow
	RHR system flow	RHR system flow
	RHR heat exchanger inlet temperature	RHR heat exchanger inlet temperature
	RHR heat exchanger outlet temperature	RHR heat exchanger outlet temperature
Rx coolant system integrity	Rx coolant loop pressure & pressurizer pressure	Rx coolant system pressure
	Hot & cold leg temp.	Rx coolant av.temp.
	Cold leg temperature	Cold leg temperature
	Rx vessel level	Rx vessel level

NOTE: Rx = reactor

CSF	Monitored Parameter	Displayed Parameter
	Pressurizer level	Pressurizer level
	Containment radiation (rad)	Containment rad
	Containment pressure	Containment pressure
	Containment sump level	Containment sump level
	SG blowdown rad	SG blowdown rad
	Condenser air ejector rad	Condenser air ejector rad
Containment Conditions	Containment pressure	Containment pressure
	Containment sump level	Containment sump level
	Containment rad	Containment rad
Radioactivity Control	Aux bldg vent stack rad	Aux bldg vent stack rad
	Containment rad	Containment rad
	SG blowdown rad	SG blowdown rad
	Condenser air ejector rad	Condenser air ejector rad

The auditors noted the following:

- o The critical safety functions (CSFs) are not explicitly depicted in the Kewaunee SPDS. CSFs were a part of the precursor Quadrex system under the heading AIDs. CSFs were dropped completely from the site-specific Kewaunee SPDS. Although AIDs exists on the TSC and EOF terminals, outside the SPDS system, it:

  - was never fully implemented.
  - is not being maintained.
  - does not adequately reflect CSF status.
  - is not a part of SPDS.
  - is not available on the control room SPDS terminal.

- o Steam generator pressure is not included as a monitored or displayed parameter representing RCS integrity in the Kewaunee control room SPDS. (SG pressure is, however, displayed on the current top level display, a display which is CSF-independent).

- o Although post-TMI main steamline radiation monitors are installed, those parameters are not available on the SPDS. (They do not read out to any computer point).

- o Only the "normal" range (1 to 10 Rem/hr) containment radiation monitors input to the SPDS. (The post TMI high containment radiation monitors, although installed, do not readout at the SPDS or at any other computer point).

- o Containment isolation and containment hydrogen concentration are not available on the SPDS.

The auditors concluded that the Kewaunee SPDS was deficient because it failed to meet NUREG-0737 Supplement 1 SPDS requirement 7, "To display, as a minimum, information sufficient to determine plant safety status with respect to five safety functions". For example,

- Critical safety functions are not explicitly displayed in the Kewaunee SPDS.

- The logic implicit in the Radioactivity Control safety function was degraded by failure to include the main steam line radiation monitor parameter.

- The logic implicit in the Containment Conditions safety function was degraded by failure to include containment hydrogen and containment isolation status.

4.2 SYSTEM DESIGN

The Kewaunee SPDS is based on Safety Assessment System (SAS) software, developed by Quadrex, running on the plant process computer. SAS software was originally designed for a four loop plant, and has been adapted for Kewaunee's two loop design. In addition, the Accident Identification and Display System (AIDS) display portion of the top level SAS display is not considered part of the Kewaunee SPDS system, and has been deleted from control room displays. The SPDS consists of three top-level displays and associated trend graphs (see Attachment 3). The plant process computer is based on two redundant Honeywell 4500 computers. SPDS displays are presented on intelligent Chromatics color graphics terminals.

4.2.1 System Description

Software. The original SAS software was developed by Quadrex and modified by Honeywell to run on the Kewaunee plant process computer. The SAS software is written in FORTRAN 77, and has been further modified by WPS computer support personnel. The major modifications to the Quadrex SAS have involved adapting it to run on the Honeywell 4500 computers being used in the Kewaunee plant, and adapting displays to reflect the Kewaunee two loop design. In addition, WPS had developed display generation software for the Chromatics CGS 7800 color graphics displays.

The AIDS display on the SAS is not considered an operational part of the SPDS. The AIDS display box in the top left corner of the SAS display is blanked out in the control room displays. However, it is shown on the displays in the TSC. In addition, the second level trend graph displays of AIDS variables are available at all stations. If these are not considered operational, and are not a part of the SPDS system, they may present a source of potential confusion, and should be removed frcm the system.

Data Refresh. Data for the chromatics displays are refreshed approximately every two seconds. This figure, however, does not account for the time taken to input the data from the sensors, and process time in the host system. In all, data displayed on the SPDS lags real time by from 5 to 10 seconds. Changes in displays on the Chromatics is quite rapid, requiring only about 2 seconds to call up new displays. These rates are satisfactory.

Hardware Architecture. The SPDS is hosted on the plant process computer system. This system is based on two Honeywell 4500 mini-computers, with one computer acting as back-up, with automatic failure detection and switchover. Almost all system hardware is similarly redundant, with back-up components switched to automatically when equipment failures are detected. The computer system and all displays operate on a dedicated power supply, with uninterruptible power supply (UPS) back-up.

There is a major flaw in system redundancy for the SPDS. To speed system re-boot time, operating system software and key data are resident in Large Core Storage (LCS) modules. Redundant LCS are provided for the plant process control system, but not for the SPDS. Failure of the single SPDS LCS would cause a loss of SPDS displays, with no back-up capability. Serious consideration should be given to providing a back-up LCS for the SPDS.

System Reliability. Only approximate reliability figures are available. None of the computer support personnel could remember a time when both SPDS displays were unavailable in the control room unless the entire host system was down. As a result, they calculated SPDS unavailability as being equal to the total time the host system was down. Only times when the system was unavailable for at least one hour were included. Between July 25, 1984 and July 30, 1986 the Honeywell was down for 157 hours (excluding downtimes of less than 1 hour). Assuming approximately 17592 hours of potential availability, this implies 0.99% system unavailability.

The SPDS appears to work with adequate reliability. However, more care should be taken by WPS in monitoring and documenting this. A centralized record should be kept indicating downtimes for the SPDS system, and for individual SPDS displays. The decision to exclude system down times of less than one hour is questionable, since such shutdowns appear to be fairly frequent (at least several times a week). All losses of SPDS displays greater than a few seconds should be logged by plant operational personnel and computer support staff.

4.2.2 Display Configuration

The audit team was tasked to confirm that the display hardware for parameters representing all five critical safety functions constituted a concise display to aid operators in determining the critical safety function status.

As indicated above, the Kewaunee SPDS does not employ the critical safety functions. Although most of the required parameters are provided, they are simply presented as time-trended values or as unprocessed/processed single parameters (e.g. narrow range steam generator level). In no case was one parameter processed with respect to another to evaluate critical safety function (e.g. if power >5% and a trip signal, then display reactivity control critical safety function red). The Kewaunee displays never relate to a critical safety function. Consequently, the auditors found the Kewaunee SPDS was deficient in that it failed to provide a concise display to aid operators in determining the critical safety function status.

In addition, the auditors noted that there was no provision to ensure that one of the two control room SPDS monitors remained on top level display at all times. Although plant personnel indicated that this was handled procedurally, no one was able to cite the specific procedure nor were the operators interviewed aware of any such restriction.

4.2.3 Data Validity

None of the shift personnel questioned were certain of the meaning of "FAIL". The question was asked directly and then the same individual was asked to explain the meaning of A train aux feed zero, B train aux feed flow FAIL. Only one of those questioned was certain whether the A train reading was valid.

The auditor noted the following apparently erroneous indications. When queried, the operators could not explain the indications but indicated most had persisted for many months:

- At 100% power, in normal lineup, SPDS indicated RHR flow ~320 GPM. RHR pumps were secured and valve lineups were normal for power operations, e.g. system flow zero. Operators disregarded the signal.

- The environment box indicated red due to high containment pressure. The signal was valid but it appeared that the algorithm set point was too low. The signal was being disregarded by the operators since it "was almost always that way".

- Reactor vessel level indicated zero and was consistently red when at 100% power. (A recent software modification was made to show void fraction with pumps running and Reactor Vessel Level Indicator System (RVLIS) level with the pumps off. Apparently, in error, the reactor vessel level was allowed to remain active with pumps on. It was not clear why level was zero. In any case, the vessel level signal was wrong, was red, and was being disregarded by the operators.

The SAS system checks data validity only when identical multiple sensors provide input to single SPDS parameters. Questionable sensor data is eliminated as long as two or more sensors remain. Questionable data is highlighted on the screen, and sensor failure messages are generated. This system is working properly. Where multiple identical sensors are not available, no validity checking is done, other than to establish whether data values exceed preassigned limits.

This system ignores two major types of potential validity checks. First, it does not compare sensors of different types which are logically related, such as wide and narrow range sensors. Second, and perhaps more importantly, it does not check the validity of sensor inputs that were aggregated by the plant process computer system. For example, if the process computer provides SAS with an average value of several sensors, SAS treats the data as if it were coming from a single sensor. Since the process computer software does no validity checking (other than ascertaining that sensor voltages are within allowable ranges) it is quite possible for faulty data from one or more sensor to be included in the averaged data provided the SAS. SAS has no way of checking this, and would present the misleading average as being accurate.

4.2.4 Maintenance and Configuration Control

WPS has established administrative software maintenance control procedures (See ACD 1.9, Attachment 4) to control, monitor and document software modifications. While these procedures do provide adequate control for administrative handling of these changes, they do not provide adequate documentation of the actual changes made to system software. The ACD 1.9 documentation does not include much of the detailed information that would be required for another programmer to understand the resulting software modifications. However, informal procedures that are currently being used by WPS computer support personnel to document modifications to the Honeywell-provided software do provide this type of information. These procedures should be formalized, and extended to cover Chromatics display generation software as well. Currently there is little documentation or configuration control on Chromatics software.

There are major gaps in SPDS system documentation. Honeywell provided little SPDS system documentation other than that provided by Quadrex for their original SAS software. In general, available SPDS software is limited to very high level system specifications and source code listings, with no documentation in between. Almost no documentation is available on Chromatics display generation software other than source code listings. Fortunately, source code listings are commented well and are reasonably easy to read, but this does not eliminate the need for adequate system documentation. Without adequate documentation, replacing existing key computer support personnel may cause serious problems in the continued maintenance of the SPDS system.

System hardware maintenance procedures appear adequate, with one exception. During routine sensor calibration checks the plant process computer system displays are checked for accuracy, but not the SPDS displays. While such checks are not always practical (since SPDS parameters are sometimes aggregations of inputs from several sensors) when possible SPDS values should be checked as well.

4.2.5 Security

SPDS system security is controlled by physically limiting system access to authorized personnel. Several modems allow telecommunications access to remote terminals. However, they are designed to allow only data display, and cannot make data changes that could alter or modify system software or data bases. While there is some possibility of unauthorized access to these ports, they do not pose a risk to the accuracy and availability of the SPDS system.

It is possible for unauthorized changes to be made to the SPDS system from terminals within the Kewaunee plant. Operations personnel can make changes to specific numbers in the system data base from SPDS terminals in the control room. This capability is under administrative control; the required keys can only be activated by unlocking the keyboard with a key held by the chief operator. This is not a serious security risk as long as tight control is maintained, and computer support personnel are aware of, and double check, any data base changes made by control room operations personnel.

Access to system programming consoles is limited only by physical security. The programming room is in a restricted access area, with open access from the TSC. Ideally, this physical security should be coupled with a password authorization system which would only allow system programmers to modify system software. While this additional measure may not be practical under the present operating system, it should be implemented if feasible.

4.2.6 Electrical Isolation

Isolation was found acceptable during an evaluation of the Kewaunee DCRDR and was so documented by an internal NRC letter dated October 24, 1984.

4.3 System Verification and Validation

Little SPDS validation work has been performed by WPS. They rely on Quadrex's earlier SAS validation work conducted at the Indian Point simulator. The system tested there was significantly different since the AIDS display was considered part of that system. WPS validation testing will not be conducted until the SPDS is available in the Kewaunee simulator, which is anticipated to take approximately one year. System validation will not be complete until the simulator-based studies have been conducted. Until validation testing is completed, the impact of removing the AIDS display from the SPDS cannot be determined and the utility of the SPDS can only be evaluated subjectively.

The SPDS system does appear to provide accurate data. Full verification testing was conducted at the Honeywell factory site. When installed in Kewaunee the new Honeywell computer system was run in parallel with the old plant process computer for a period of time. Parameters from the two systems were compared to establish the validity of the new system. While it is not clear to the audit team whether this testing extended to the SPDS outputs, subsequent tests which injected test values into the plant process computer datasets and checked these against SPDS outputs indicated data was being properly processed by the SPDS.

The only weakness in the WPS verification testing is the lack of complete end-to-end testing, where a signal injected into a sensor is checked against the parameter displayed on the SPDS. As discussed above under data validity, where possible, this type of end-to-end testing should be incorporated into standard procedures for sensor calibration.

4.4 Human Factors Engineering

The SPDS is defined by the WPS as the three top-level displays and the trend graph displays (see Attachment 3). The three top-level displays include the Normal, Heatup/Cooldown and Cold Shutdown displays. The SPDS is a segment of the overall Safety Assessment System (SAS).

The SAS, as originally designed, was intended to function as an integrated system to indicate off-normal conditions on the Safety System Readiness Monitor, Safety System Performance Monitor, and Critical Safety Function Monitor. In addition, the Accident Identification and Display System was designed to aid the operator in the recognition of four events: loss of coolant accident (LOCA), steam generator tube rupture (SGTR), loss of secondary coolant (LOSC) and inadequate core cooling (ICC). These systems were designed to meet the integrated functional requirements of SAS. However, these systems were eliminated. This significantly reduced the value of the display system as an SPDS.

The top-level displays originally included three major areas, SPDS, Message Area and AIDS. AIDS is no longer part of the top-level displays and there is no integration between the display system and the four events, LOCA, SGTR, LOSC, and ICC, but the individual supporting AIDS displays are still in the system. In addition, the elimination of the critical safety function monitoring system which was intended to monitor the status of the CSF trees in the EOPs further reduces the SPDS integration with emergency operations. The operator is left with a display of unprocessed parameters which are the same as what he has on the control panels. Therefore, the

SPDS does not add to the operator's tools for identifying and coping with accidents.

Even though the AIDS top-level display was removed from the control room, the AIDS function keyboard was not removed. This allows the operators to access the individual LOCA, SGTR, LOSC, and ICC displays. Since these displays have been left in the system, even though they are no longer part of the SAS or SPDS, there is potential for misleading the operators. The AIDS function keyboard in the control room should either be covered or removed to prevent operator access.

The trending arrows on the top-level displays provide misleading trend direction information. This is because the trend arrows are designed to respond to a small set of signals. In fact, the trend arrow can be pointing in the opposite direction from the parameter trend. The most effective way to evaluate trend is to call up the appropriate trend graph. The trend arrows should correctly reflect trend or be removed from the top-level displays.

With regard to workspace location, the SPDS is located conveniently to control room operators. However, the lettering on the displays is small and the displays are not bright. This means that the user must be very near the display in order to use the SPDS information.

In summary, there are several significant man-machine interface problems with the Kewaunee SPDS. The top-level displays and trend graphs only partially fulfill the functional requirements of the original SAS concept. The elimination of the AIDS and critical function monitoring system makes it difficult for the operator to relate unprocessed top-level parameter information to the critical safety functions and emergency operations. Leaving the AIDS function keys on the SPDS keyboard allows operator access to the AIDS displays which are no longer maintained as part of SPDS. The trending arrows on the top-level displays are misleading. The workspace location is convenient, but the displays are dim and the print is small.

4.5 Use of SPDS in Operation

One purpose of the audit was to judge whether the SPDS satisfies its intended purpose of aiding operators in "rapidly and reliably determining the safety status of the plant" and "in ...assessing whether abnormal conditions warrant corrective actions by operators to avoid a degraded core". The audit team assessment was based on: (1) demonstrated operation of the SPDS in the control room with the plant in operation, (2) interviews with licensed operators and an STA held at the TSC SPDS console, and (3) discussions with training and computer system personnel.

With very limited exceptions, the Kewaunee SPDS is simply a CRT display of information which is already available elsewhere in the control room. Critical safety function status is not explicitly displayed nor has the SPDS been coordinated with the symptomatic emergency operating procedures as is required by NUREG-0737 Supplement 1. Although the systems personnel indicated that signals could be blocked from the SPDS during instrument or loop surveillance testing, the operators were not aware of how this was accomplished and, as a result, test signals routinely pass into the SPDS until the tested signal is excluded as it goes "out of range." The result is degraded SPDS data. Long term tolerance of aberrations (such as reactor vessel level zero and red, RHR flow -320 GPM, etc.) have degraded operator confidence and acceptance of the system. The operators, when questioned, were quick to describe the system as "useful." However, except for startup steam generator level control trending, the operators appeared to find little day-to-day use for the system. Operator training was accomplished on a one-time basis 18 months ago (see Attachment 5). No retraining is available or required. The absence of an SPDS in the simulator precludes operator usage under simulated accident conditions. No one is assigned responsibility to monitor the SPDS under normal or emergency conditions nor is the role of the SPDS during emergency operations defined. No SPDS users manual or operating procedures are available; no reference material is available for operators trying to research a question on SPDS operation.

Due to limited time and the absence of system documentation, the audit team was unable to verify that the sampling rate for each CSF/Parameter was such that there was no significant loss of information. However, in general, response times appeared to be adequate; display screen update rate was excellent.

Operators indicated that the system was reliable and their estimates of system downtime tended to support the 157 hours downtime in 18 months as furnished by the utility.

In summary, the Kewaunee SPDS serves no purpose except that it addresses an NRC requirement, inadequately. Reviewing NRC IE Inspection Notice 86-10 (Reference 9), the audit team concluded that the Kewaunee SPDS suffered the following deficiencies noted there:

- Display of unreliable or invalid data
- Poor acceptance by operators
- Failure of management to integrate the SPDS into the operational environment
- Inadequate documentation

5.0 CONCLUSIONS

The post-implementation audit of the Kewaunee SPDS was conducted October 7 and 8, 1986. During the audit, the NRC audit team identified the following set of specific concerns associated with the SPDS. The conclusions are presented in terms of the eight Supplement 1 to NUREG-0737 requirements.

1. The SPDS does not present a concise display of critical plant variables because the parameters selected for SPDS display do not include:

    - o Main steamline radiation
    - o Containment isolation valve position
    - o Containment hydrogen concentration

2. The SPDS is located convenient to control room operators.

3. The SPDS is not continuously displayed. The licensee has no provision for ensuring that one of the two control room SPDS monitors continuously displays a top-level display.

4. The SPDS is reliable in the sense that the computer and the displays function most of the time. However, at the time of the audit, the SPDS displayed several erroneous indications, including incorrect residual heat removal flow, environment box incorrectly colored red, and reactor vessel level indicating zero when operating at 100% power. Some of the erroneous indications had been present for lengthy periods prior to the audit.

5. The SPDS is suitably isolated from electrical and electronic interference with safety systems. (This was determined previous to and independently of the audit.)

6. The SPDS has a number of human engineering discrepancies, listed below:

   a. The Accident Identification and Display System (AIDS) display has been removed from the top-level display box, but the AIDS supporting displays are still available on the SPDS. This could lead to confusion.

   b. The Accident Identification and Display System function keys remain active on the SPDS keyboard, even though this system is no longer used.

   c. The trend arrows on the top-level displays are misleading.

   d. The top-level displays in the control room are not bright.

   e. The top-level displays use small lettering.

7. The SPDS does not display information sufficient to determine the following critical safety functions:

a. Radioactivity control variables are not comprehensive because main steamline radiation is not included in the SPDS.

b. Containment conditions variables are not comprehensive because containment isolation valve position and containment hydrogen are not included in the SPDS.

c. The SPDS does not directly display the critical safety functions required by Supplement 1 to NUREG-0737 or identified in the emergency operating procedure critical safety function trees.

8. Procedures and operator training addressing action with and without SPDS are inadequate.

a. There is no operating manual for the SPDS.

b. One-time training on SPDS is inadequate to keep operators current on its use.

c. The SPDS is not coordinated with emergency operating procedures or integrated into emergency operations.

d. No operator (reactor operator, shift supervisor, or shift technical advisor) has been assigned the task of using SPDS information in emergencies.

9. Other Conclusions:

a. Since the process computer software does no validity checking (other than ascertaining that sensor voltages are within allowable ranges), it is possible for faulty data from one or more sensors to be included in the averaged data provided by SAS.

b. Only informal procedures are currently being used by the licensee's computer support personnel to document modifications to SPDS software.

c. There are major gaps in SPDS system documentation.

d. A weakness in the WPS verification testing is the lack of complete end-to-end testing, where a signal injected into a sensor is checked against the parameter.

e. Failure of a single large core storage (LCS) module would cause loss of SPDS displays.

Based on the concerns identified above, the NRC audit team concluded that the Kewaunee SPDS does not meet the requirements of Supplement 1 to NUREG-0737.

REFERENCES

1. NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," USNRC, Washington, D.C., December 1982.

2. Kewaunee Nuclear Power Plant Safety Parameter Display System Safety Analysis Report, Wisconsin Public Service Corporation, September 2, 1983.

3. Additional Information on the Kewaunee Nuclear Power Plant Safety Parameter Display System, Wisconsin Public Service Corporation, August 1, 1984.

4. Letter from D.C. Hintz (WPS), to: S.A. Varga (NRC), Subject: Additional Information on Kewaunee Nuclear Power Plant Safety Parameter Display System, Wisconsin Public Service Corporation, April 26, 1985.

5. Status of Kewaunee Safety Parameter Display System, Wisconsin Public Service Corporation, July 14, 1986.

6. Letter from D.C. Hintz (WPS), to: Morton B. Fairtile (NRC), Subject: Additional Information on SPDS, Wisconsin Public Service Corporation, August 21, 1986.

7. NUREG-0800, Standard Review Plan for Review of Safety Analysis Reports for Nuclear Power Plants, Section 18.2, Rev. 0, Safety Parameter Display System (SPDS), Appendix A to SRP Section 18.2, NRC, November 1984.

8. NUREG-0700, Guidelines for Control Room Design Reviews, NRC, September 1981.

9. IE Information Notice No. 86-10: Safety Parameter Display System Malfunctions, USNRC, February 13, 1986.

ATTACHMENT 1

LIST OF MEETING ATTENDEES

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October 7, 1986

M. A. Archer	SAIC/NRC	
G. R. Bryan	NRC (Comex)	
J. DeBor	SAIC/NRC	
R. J. Eckenrode	NRC/DPLA/PAEI	SPDS MPA Mgr
M. B. Fairtile	NRC/DPLA	Proj. Manager
S. A. Gunn	WPSC	Nuc. Technical Review Supv
D. S. Nalepka	WPSC	Nuclear Licensing Projects Supervisor
R. L. Nelson	USNRC	Senior Resident Inspector
S. N. Saba	NRC/DPLA/PAEI	CR and SPDS Reviewer
J. R. Tyrrell	WPSC	Electrical Design Engineer
J. J. Wallace	WPSC	Nuc. Computer Supervisor

ATTACHMENT 2

LICENSEE'S RESPONSE TO TEN NRC SPDS QUESTIONS

NRC Question #1

Date SPDS was declared operational. Was this in accordance with issued orders or commitment dates?

WPSC Response

Our letter dated April 26, 1985 (reference 2) informed you that the SPDS was implemented in April 1985 prior to the completion of the 1985 refueling outage. Implementation was in accordance with the NRC confirmatory order issued by letter dated June 12, 1984 (reference 3).

NRC Question #2

Date verification and validation program on SPDS was completed?

WPSC Response

Verification and validation on the Kewaunee SPDS was conducted in the following logical sequence.

1) The testing of the generic Safety Assessment System (SAS) performed at the Indian Point simulator "proved the concept" of SAS as an operator aid which meets the guidelines of NUREG-0696 and subsequent NUREG-0737, Supplement 1.

2) The WPSC installation of the SPDS portion of SAS was technically faithful to the generic system, departing only where plant design dictated variations (e.g., 2 loops vs. 4 loops).

3) Verification of the installation occurred in two phases: the determination that field inputs (pressures, flows, etc.) were accurately processed into the plant process computer system (PPCS); and second, the verification that

the SAS specific software processed the data base values properly. The methods used for those two phases are as follows:

a) Verification of accurate field input conversion was performed on each process signal during the installation process by comparing the new computer values against the old computer values. (Both the new and old computers were operating concurrently during the installation. The procedure for transferring points required verification between the two systems on a per point basis.)

b) Verification of proper SPDS manipulation was performed by a separate procedure, administered by the Wisconsin Public Service Corporation (WPSC) Construction group. This procedure, which parallels the guidelines of NSAC-39 "Verification and Validation of Safety Parameter Display Systems," was completed in May, 1985.

4) Modifications to SPDS have not caused a repeat of the last step (3b) as no major modifications have been performed. Simple signal additions (e.g., addition of auxiliary feedwater flow) are verified similar to 3a above.

5) Final plant specific validation of SPDS cannot occur until a full simulation of SAS on the KNPP Simulator is completed.

Based upon the above discussion, May, 1985 may be considered the date the initial Validation and Verification was completed. It is recognized that outstanding discrepancies exist, and that plant specific accident validation has yet to be performed.

NRC Question #3

Were operators trained and are procedures available for using SPDS?

WPSC Response

During the spring of 1985, the licensed operators received formal training regarding the capabilities and use of the SPDS. The operation of the SPDS keyboard is very logical and simple to use; therefore, initial training describing the options available to the SPDS user is sufficient. Formal procedures on the use of the SPDS are not available nor are they needed.

NRC Question #4

Have you reviewed your SPDS against the problems identified in IE Information Notice 86-10? Do you have similar problems?

WPSC Response

The above referenced Information Notice was reviewed as part of the WPSC "Operating Experience Assessment" (OEA) program. The portion of the review which has been completed concluded that adequate programs and procedures are in place to ensure that the concerns expressed in the notice are not present in the WPSC SPDS program. The items not covered during the OEA review, which included training, operator acceptance, and management support of the system are being reviewed. Some preliminary information regarding the operator acceptance of the system is provided herein with the responses to questions 3, 9 and 10.

NRC Question #5

What is the operational availability of your SPDS?

WPSC Response

A simple view of the SPDS includes two control room display units and the host (Honeywell) computer system. Unavailability of the host usually renders the SPDS system out of service. A failure of either SPDS unit has no effect on SPDS availability since the control room units are redundant.

The following periods of unavailability have been established by a preliminary review of maintenance records between July 25, 1984 and July 30, 1986.

SAS Unit 2: 100 hours down

875 hours optional shutdown due to high frequency noise (annoyance)

SAS Unit 3: 172 hours down

Honeywell (Host): 157 hours down

The following additional information is also available:

1) Based upon personnel observations, both SAS units have not failed at the same time.

2) Honeywell downtime may have overlapped SAS unit down time.

3) Downtime of Honeywell caused by system initialization procedures as part of general system maintenance, or other quick (less than one hour) outages are not included in this total.

Since SPDS availability is satisfied by one operable unit in the control room, the total unavailability for the time period was 157 hours or less than 1%.

NRC Question #6

Is the SPDS incorporated in the Emergency Procedures?

WPSC Response

The SPDS is intended as an aid to the operators to assist in accident mitigation and recovery. The Emergency Operating Procedures (EOPs) contain information which must be followed by the operators in mitigating an accident. Inclusion of references to the SPDS in the EOPs is not necessary and could adversely affect the implementation of the procedures by adding unnecessary information to the procedures. While it is important that the operator realize that the SPDS is available as an aid, inclusion of reference to the SPDS in EOPs is not warranted.

NRC Question #7

Who is the primary user of the SPDS?

WPSC Response

A survey of the Control Room Operations personnel and the Shift Technical Advisors (STA) was conducted to determine the primary user of the SPDS. The majority of the respondents indicated that the Control Room Operators are the primary users during normal operating conditions. During emergency conditions the respondents indicated that the Control Room Operators, the Control Room Supervisors, the Shift Supervisor and the STA could all be users of the SPDS. Use of the system under upset conditions will be better defined after implementation on the plant specific simulator.

NRC Question #8

Have modifications been made to the SPDS since it was declared operational?

WPSC Response

Some modifications have been made to the SPDS since it was declared operational. Some additional data points (i.e., auxiliary feedwater flow, wide range containment pressure and wide range containment sump level) have been added and some minor software modifications have been implemented.

NRC Question #9

Does the operational staff believe that the SPDS makes the operation job easier?

WPSC Response

Most of the operators agree that the SPDS does make the job easier during normal operations. The trending capabilities during unit startup and shutdown seem to be most useful.

Many operators believe that the SPDS will make their job easier during emergency conditions as it will provide an additional aid for postaccident recovery. However, many operators indicated that the control board instrumentation will remain the primary source of information during implementation of the EOPs.

NRC Question #10

To what extent are operations staff using or relying on information provided by the SPDS?

WPSC Response

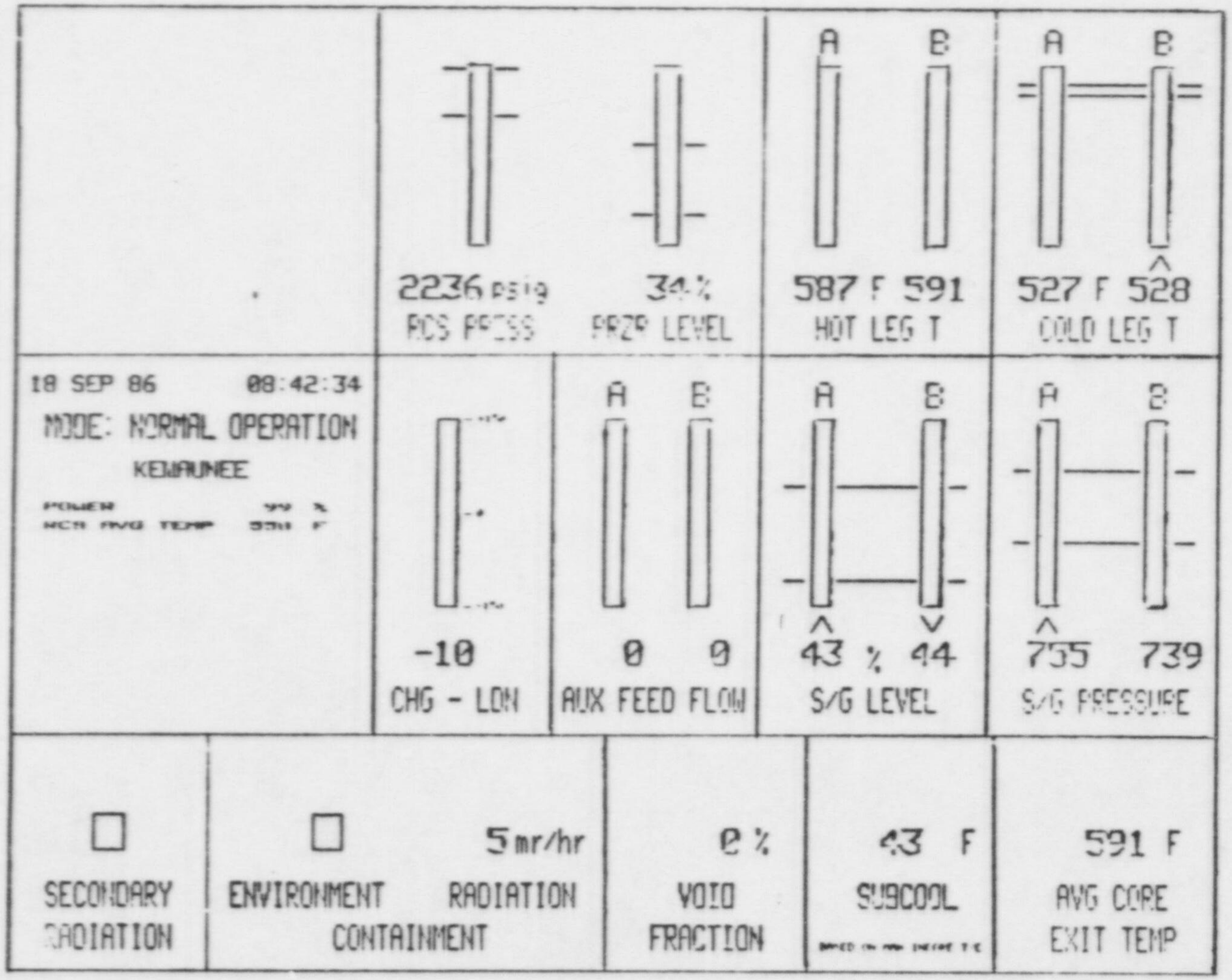
The responses to the survey indicated that many operators use the SPDS frequently while others tend not to use the system at all. The capability to trend parameters during normal and emergency conditions was mentioned most often by the operators who use the system.

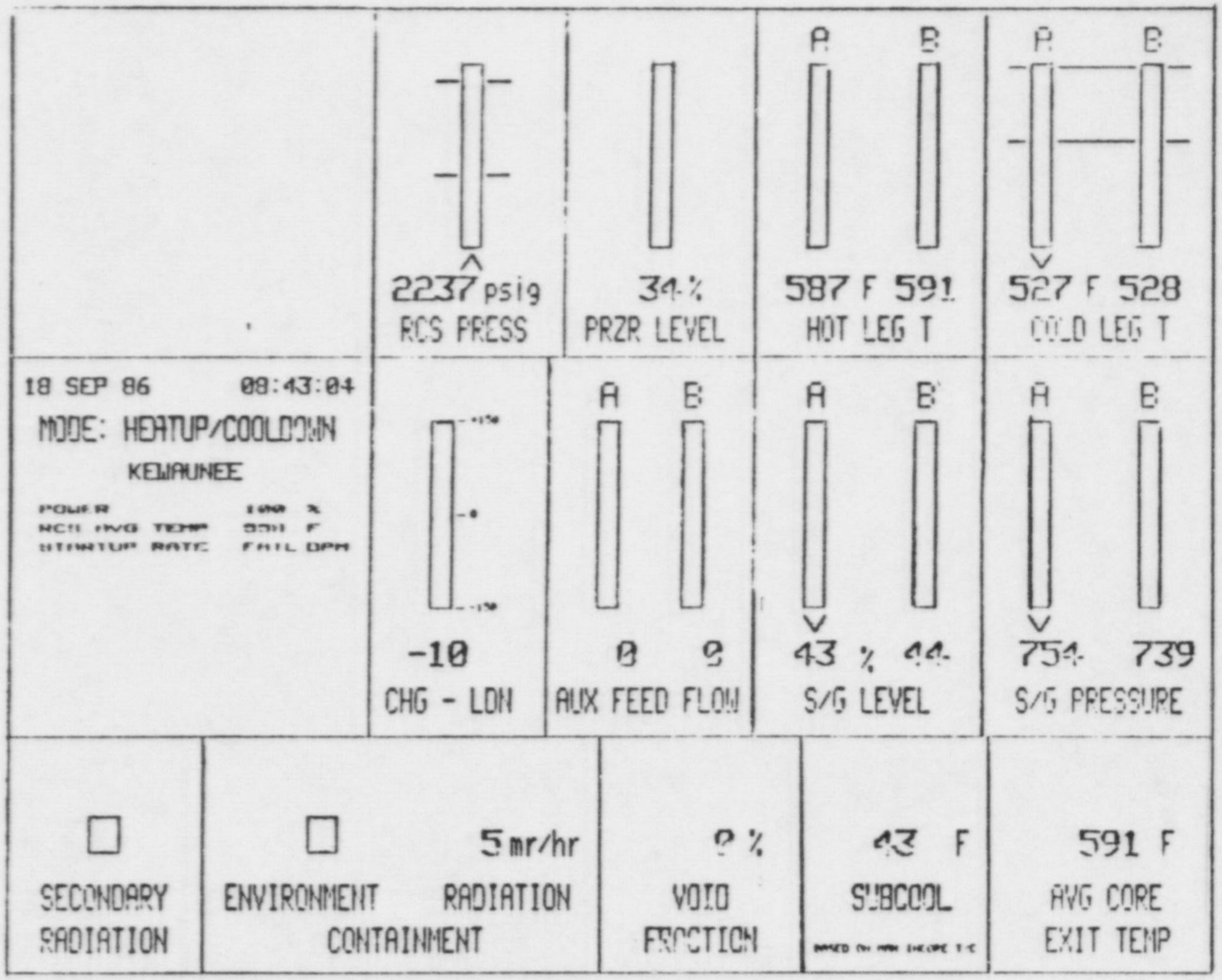
Many operators indicated that it is difficult to anticipate how much the SPDS would be used during emergency conditions. The control board instrumentation would remain the primary indication during procedure implementation and diagnosis, however, it is recognized that the SPDS will provide a quick look summary that can periodically be referenced and has proven useful during normal transient evolution (e.g., heatup and cooldown).

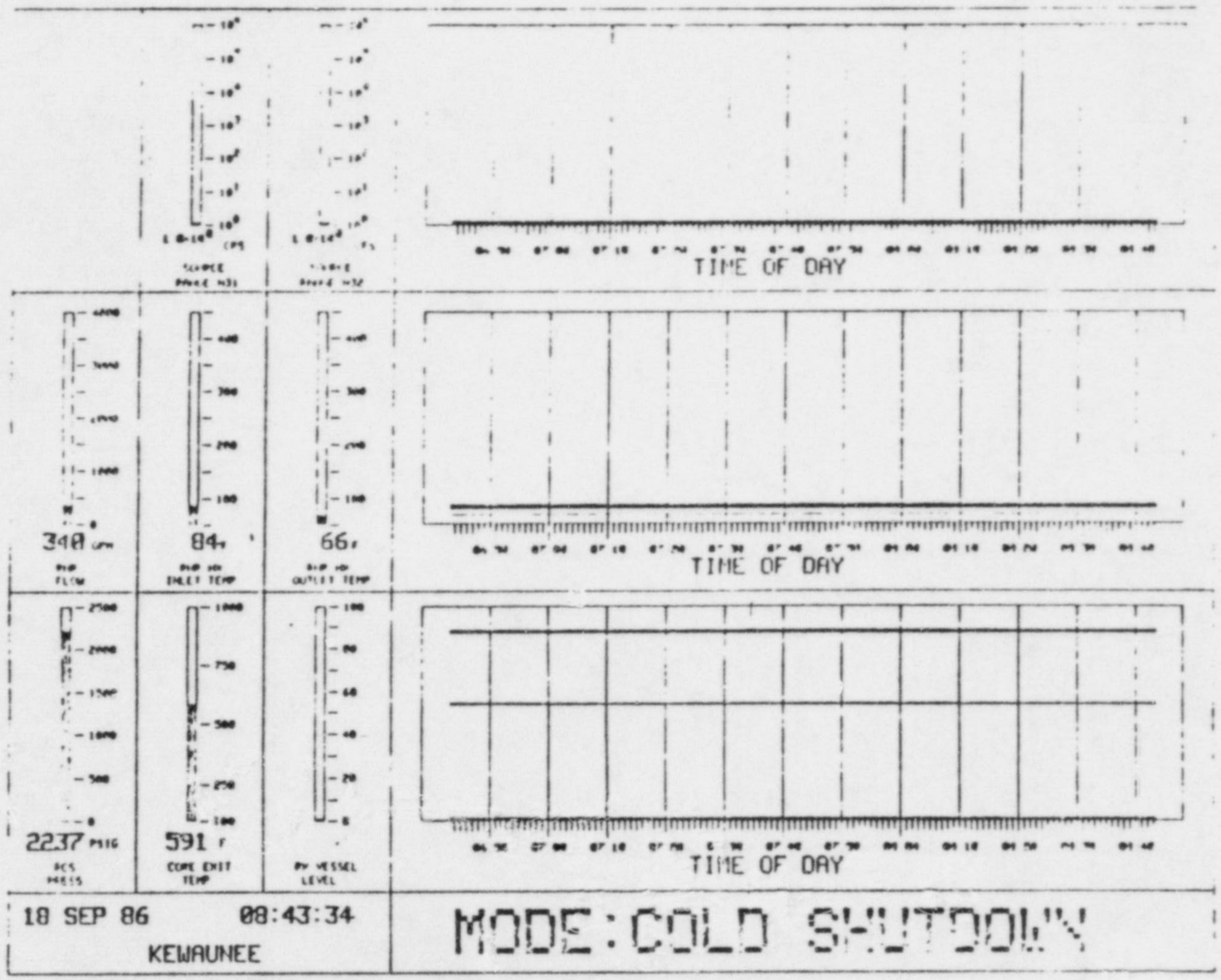
Since the SPDS is not currently available on the KNPP Simulator, many operators indicated that training on the Simulator using SPDS for real time evaluations will help enhance the operators confidence in the system.

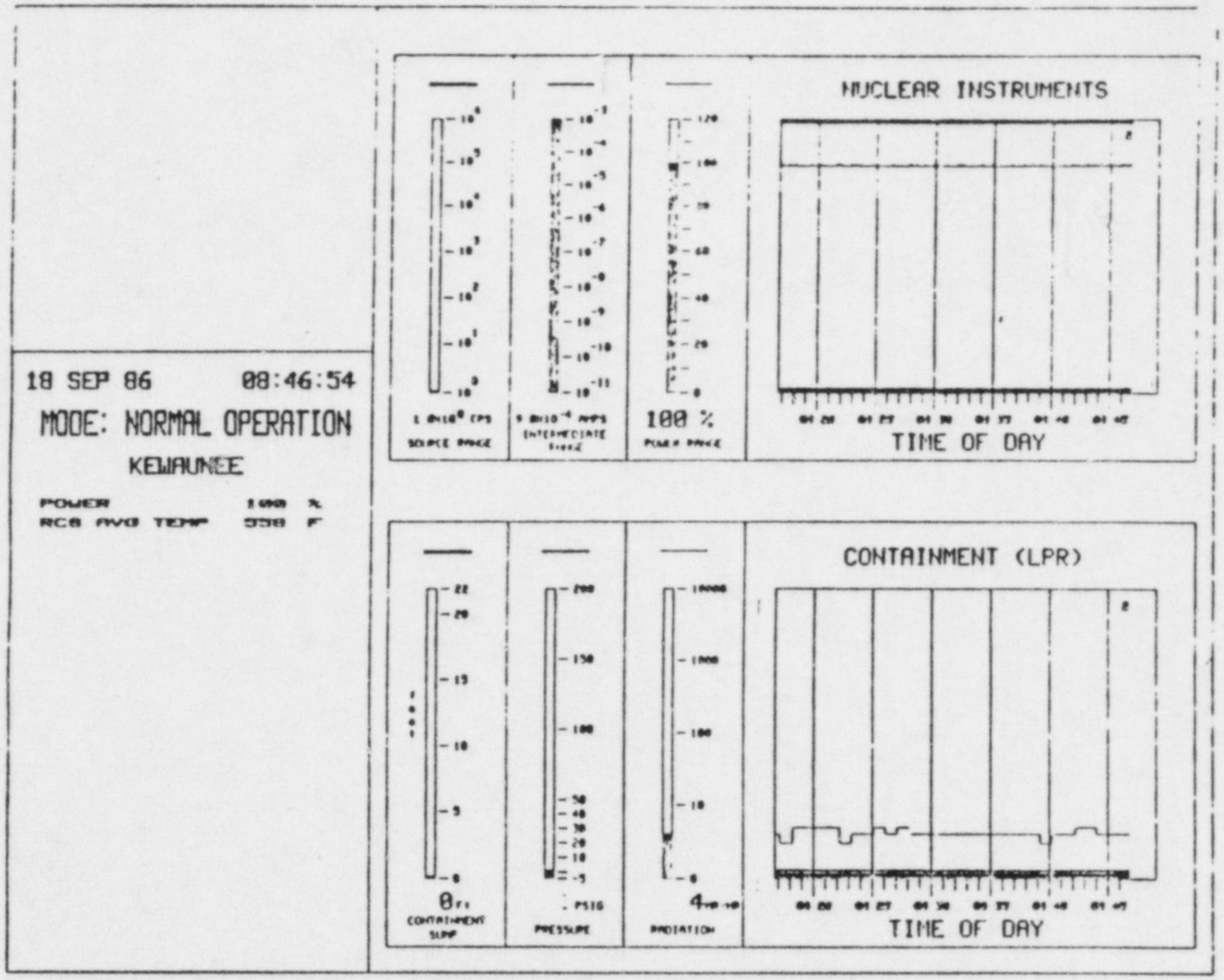
ATTACHMENT 3

SPDS TOP LEVEL AND TREND DISPLAYS


2236 psig
RCS PRESS
34%
PRZR LEVEL
A B
587 F 591
HOT LEG T
A B
527 F 528
COLD LEG T
18 SEP 86 08:42:34
MODE: NORMAL OPERATION
KEWAUNEE
-10
CHG - LDN
A B
0 0
AUX FEED FLOW
A B
43 % 44
S/G LEVEL
A B
755 739
S/G PRESSURE
SECONDARY
RADIATION
5 mr/hr
ENVIRONMENT
RADIATION
CONTAINMENT
0 %
VOID
FRACTION
43 F
SUBCOOL
591 F
AVG CORE
EXIT TEMP


A B
A B
2237 psig
RCS PRESS
34%
PRZR LEVEL
587 F 591
HOT LEG T
527 F 528
COLD LEG T
18 SEP 86 08:43:04
MODE: HEATUP/COOLDOWN
KEWAUNEE
A B
A B
A B
-10
CHG - LDN
0 0
AUX FEED FLOW
43 % 44
S/G LEVEL
754 739
S/G PRESSURE
5 mr/hr
SECONDARY
RADIATION
ENVIRONMENT RADIATION
CONTAINMENT
0 %
VOID
FRACTION
43 F
SUBCOOL
591 F
AVG CORE
EXIT TEMP


TIME OF DAY
340 GPM
84 F
66 F
TIME OF DAY
2237 PSIG
591 F
TIME OF DAY
18 SEP 86
08:43:34
KEWAUNEE
MODE:COLD SHUTDOWN

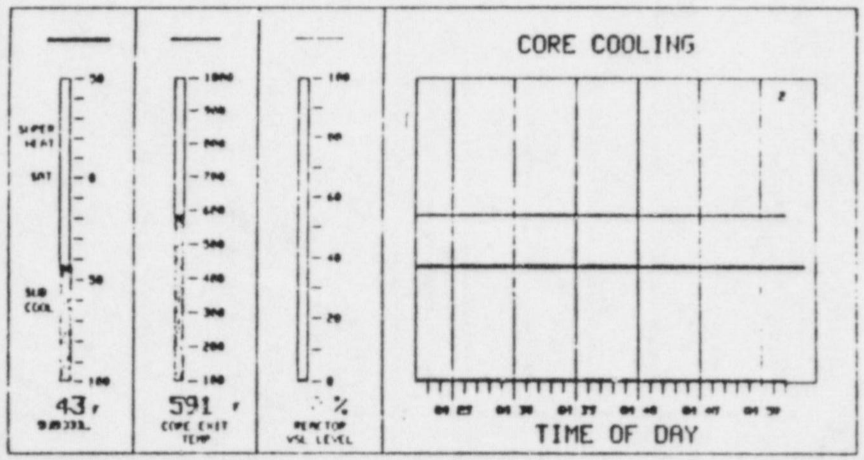

18 SEP 86
08:46:54
MODE: NORMAL OPERATION
KEWAUNEE
POWER 100 %
RCS AVG TEMP 558 F
NUCLEAR INSTRUMENTS
100 %
TIME OF DAY
CONTAINMENT (LPR)
CONTAINMENT SUMP
PRESSURE
PSIG
RADIATION
TIME OF DAY

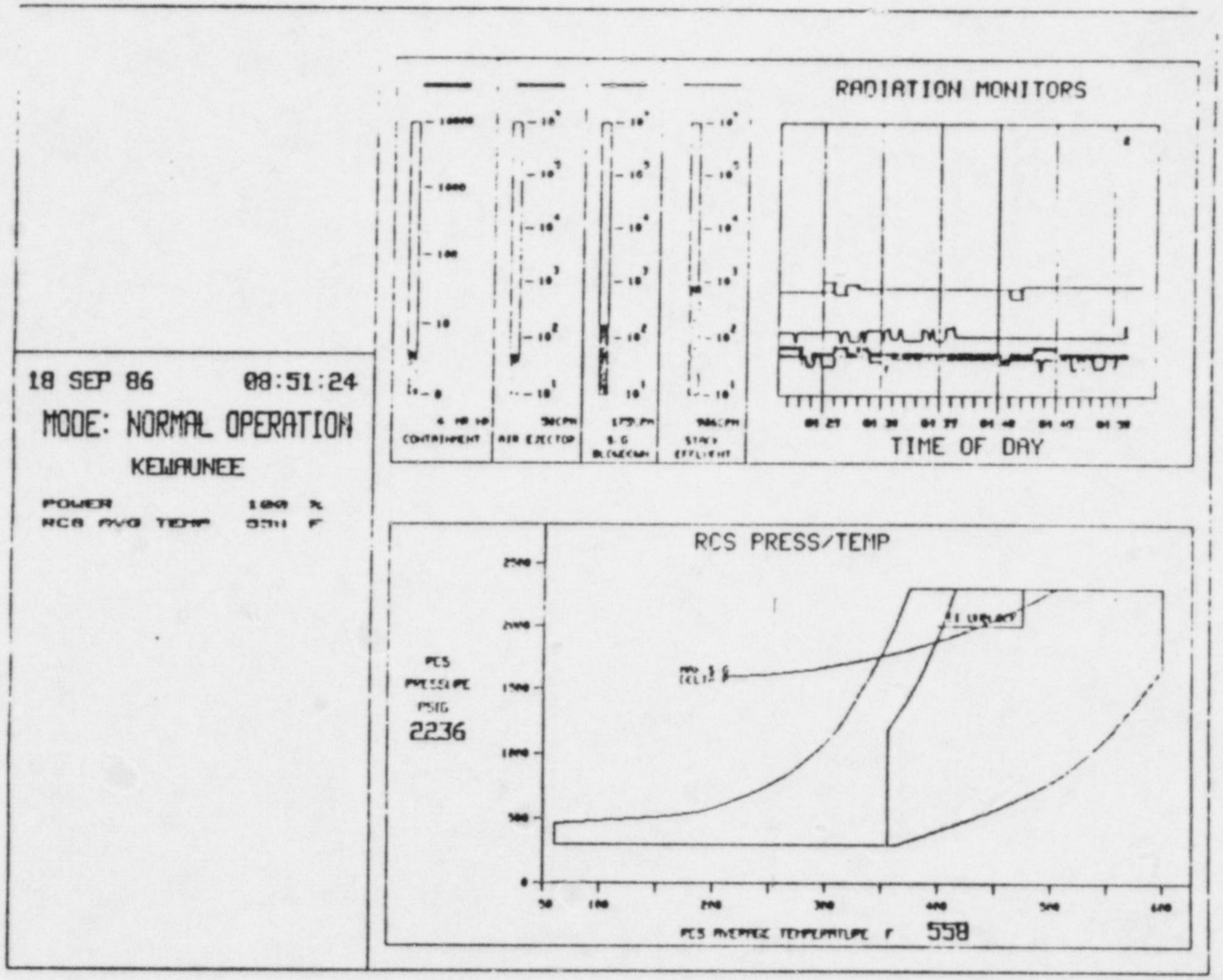
18 SEP 86 08:52:10

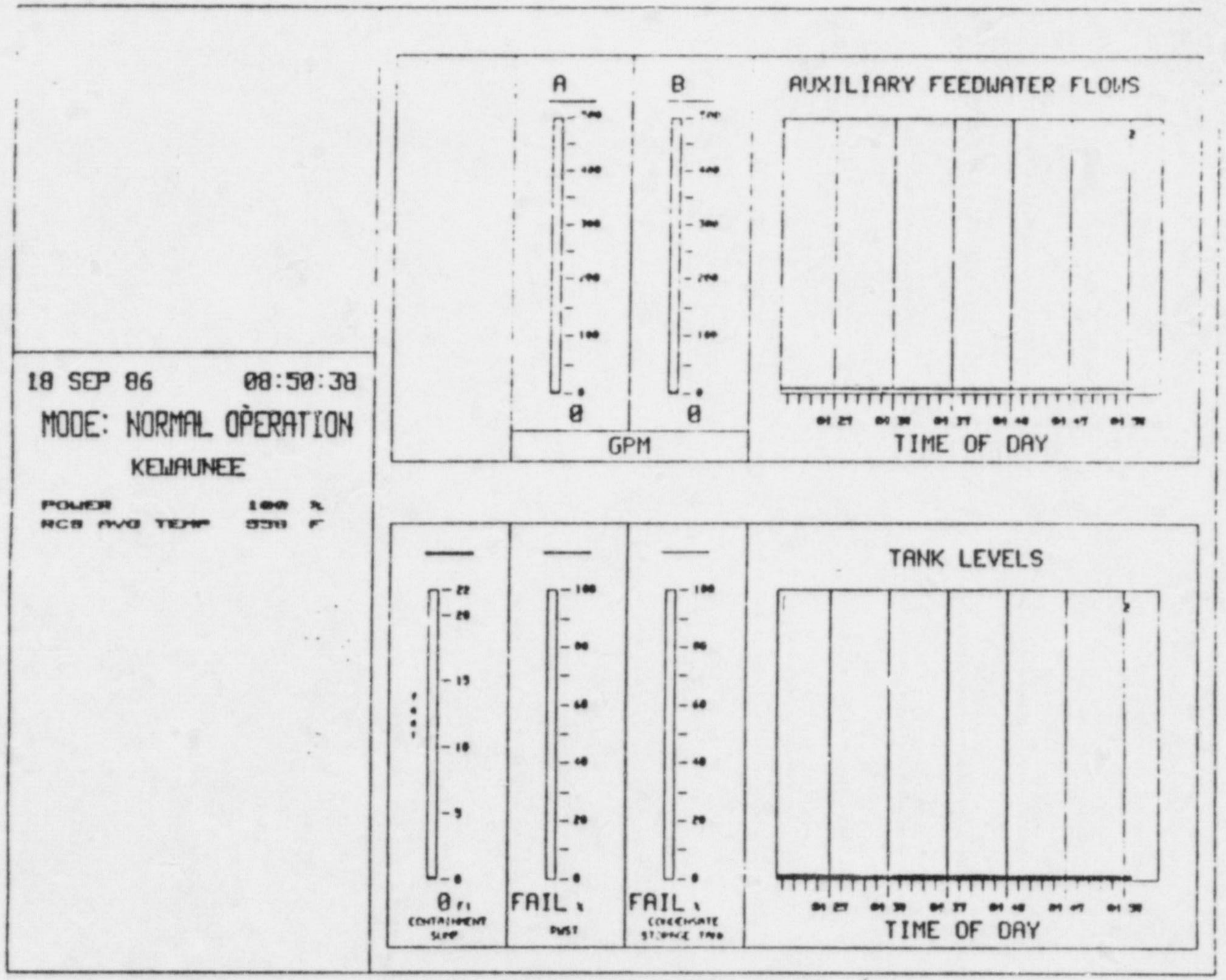
MODE: NORMAL OPERATION

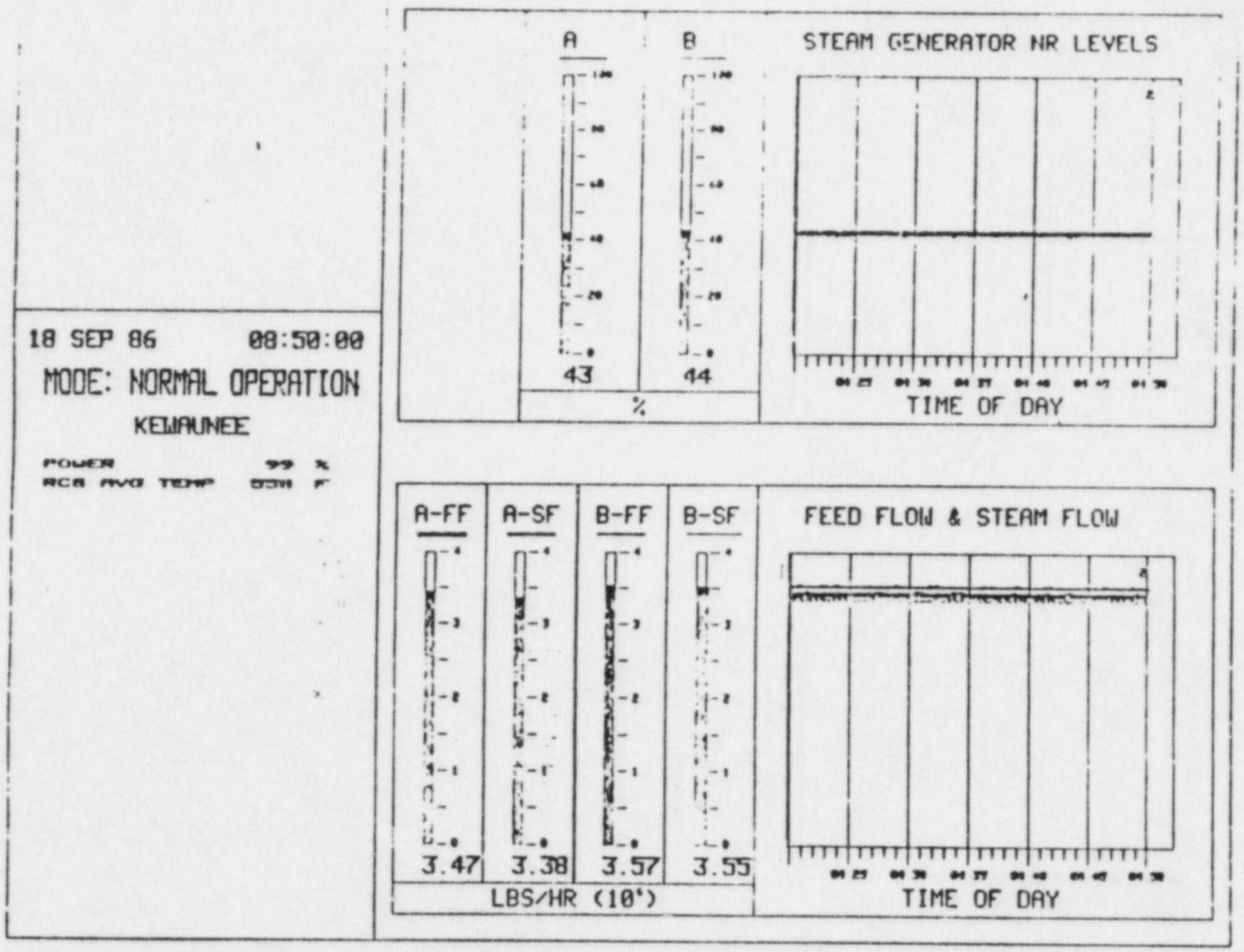
KEWAUNEE

POWER 100 %
RCS AVG TEMP 558 F

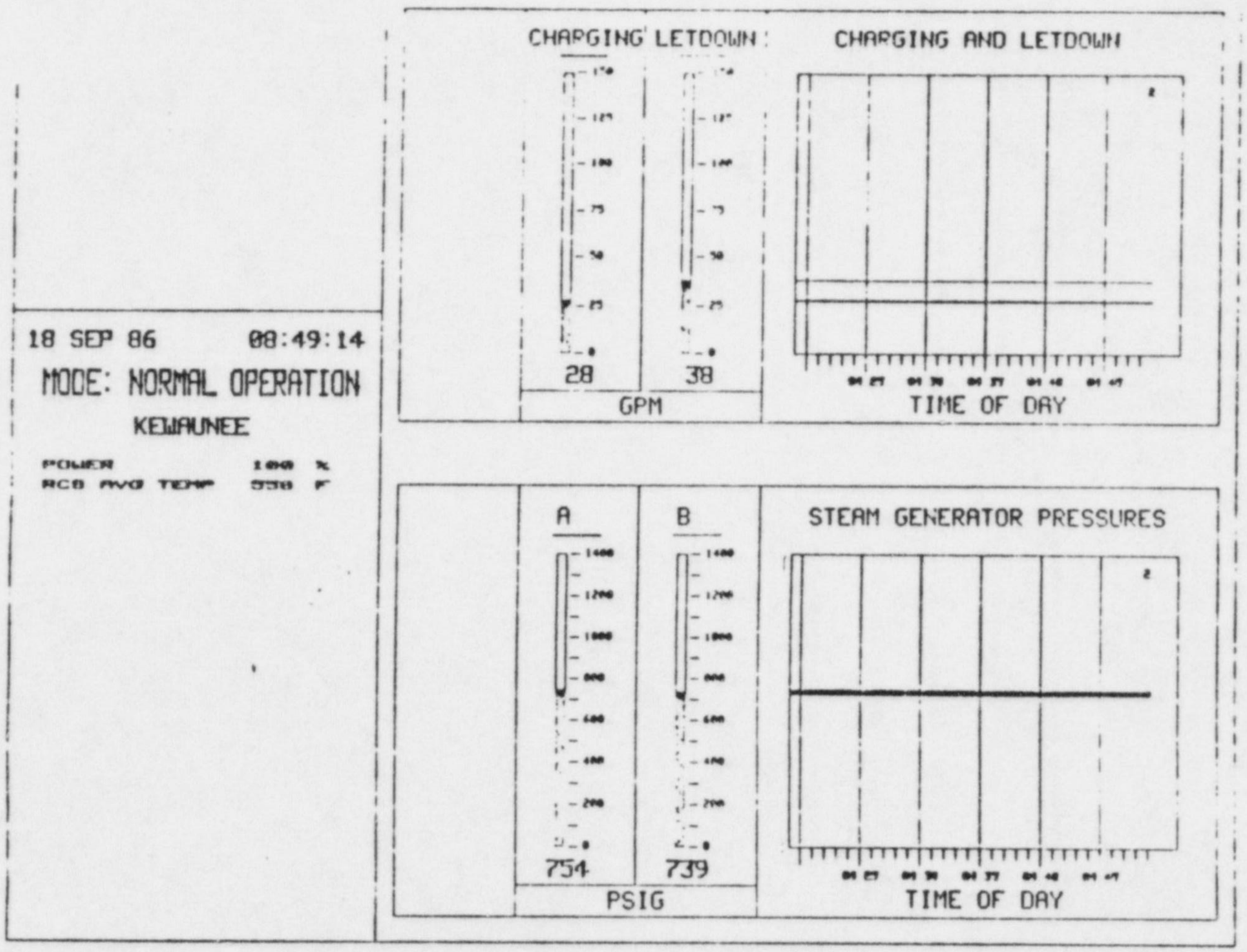



18 SEP 86
08:51:24
MODE: NORMAL OPERATION
KEWAUNEE
POWER
RADIATION MONITORS
CONTAINMENT
AIR EJECTOR
TIME OF DAY
RCS PRESS/TEMP
RCS
PRESSURE
PSIG
2236
558

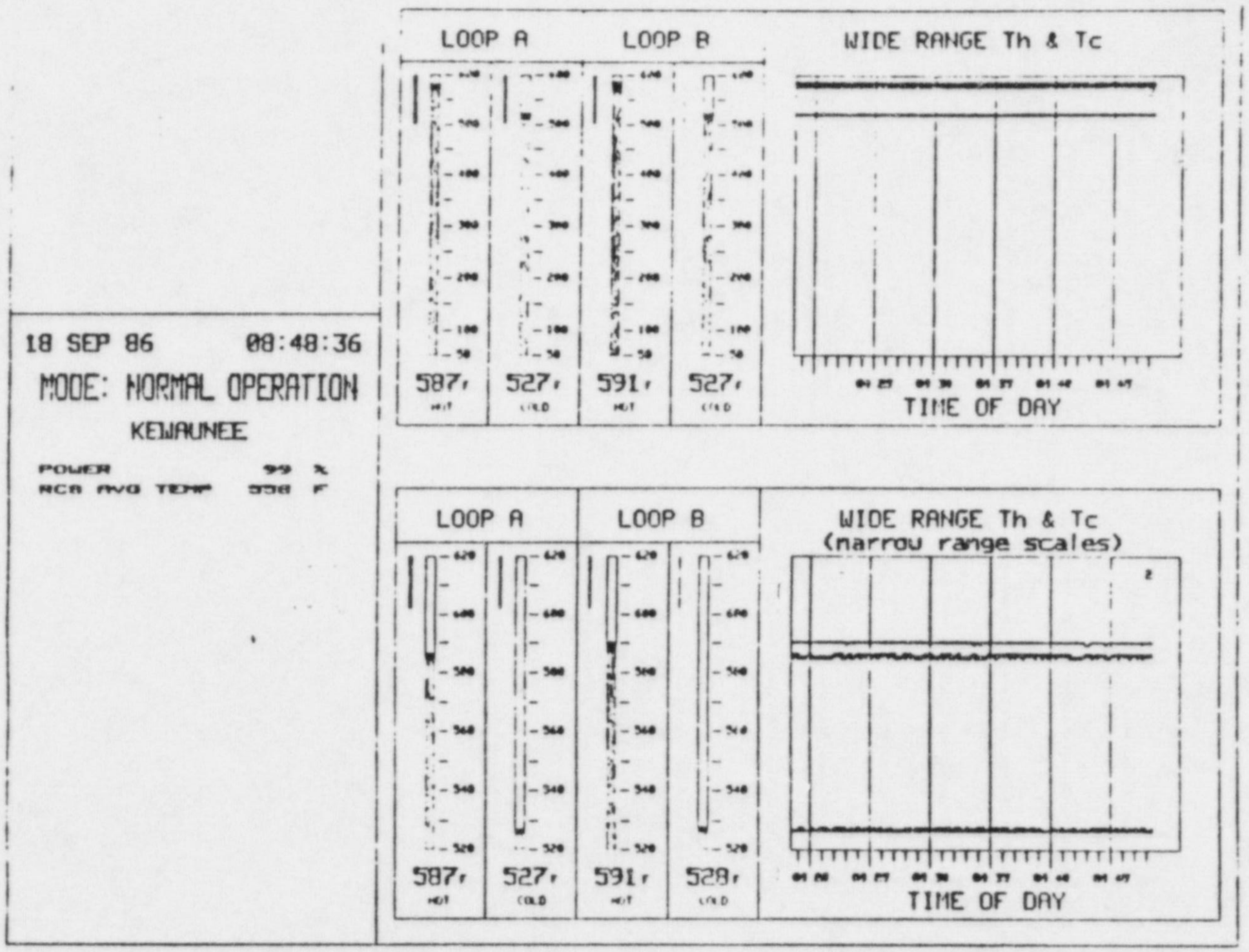

18 SEP 86
08:50:38
MODE: NORMAL OPERATION
KEWAUNEE
POWER
100 %
RCS AVG TEMP
A
B
AUXILIARY FEEDWATER FLOWS
0
0
GPM
TIME OF DAY
TANK LEVELS
0 FT
CONTAINMENT SUMP
FAIL
RWST
FAIL
CONDENSATE STORAGE TNK
TIME OF DAY

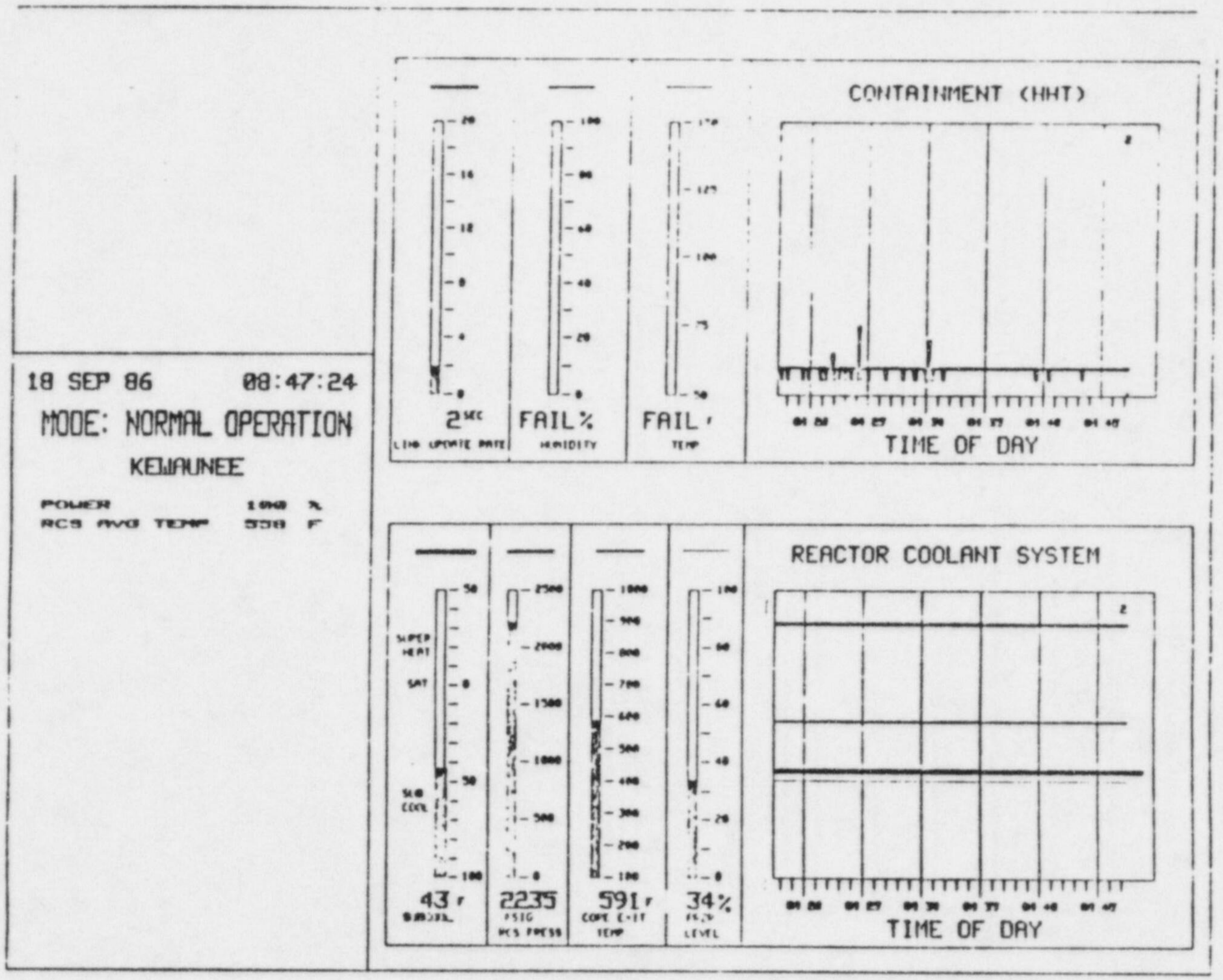


18 SEP 86
08:50:00
MODE: NORMAL OPERATION
KEWAUNEE
POWER 99 %
RCS AVG TEMP
A
B
STEAM GENERATOR NR LEVELS
43
44
%
TIME OF DAY
A-FF
A-SF
B-FF
B-SF
FEED FLOW & STEAM FLOW
3.47
3.38
3.57
3.55
LBS/HR (10⁶)
TIME OF DAY



18 SEP 86
08:49:14
MODE: NORMAL OPERATION
KEWAUNEE
POWER 100 %
RCS AVG TEMP 558 F
CHARGING LETDOWN
28
38
GPM
CHARGING AND LETDOWN
TIME OF DAY
A
B
754
739
PSIG
STEAM GENERATOR PRESSURES
TIME OF DAY


18 SEP 86 08:48:36
MODE: NORMAL OPERATION
KEWAUNEE
POWER 99 %
RCS AVG TEMP 558 F
LOOP A
LOOP B
WIDE RANGE Th & Tc
587F HOT
527F COLD
591F HOT
527F COLD
TIME OF DAY
LOOP A
LOOP B
WIDE RANGE Th & Tc
(narrow range scales)
587F HOT
527F COLD
591F HOT
528F COLD
TIME OF DAY


18 SEP 86
08:47:24
MODE: NORMAL OPERATION
KEWAUNEE
CONTAINMENT (HHT)
2 SEC
FAIL %
FAIL
TIME OF DAY
REACTOR COOLANT SYSTEM
43 F
2235
591 F
34%
TIME OF DAY

ATTACHMENT 4

ADMINISTRATIVE CONTROL DIRECTIVE (ACD 1.9)
ADMINISTRATIVE SOFTWARE MAINTENANCE CONTROL PROCEDURES

WISCONSIN PUBLIC SERVICE CORPORATION Kewaunee Nuclear Power Plant ADMINISTRATIVE CONTROL DIRECTIVE	NO. ACD 1.9 REV. A TITLE: Software Maintenance Control DATE APR 4 1986 PAGE 1 of 11
REVIEWED BY [signature] Department Head REVIEWED BY Dave Berg Superintendent-Plant QC	APPROVED BY [signature]

1.0 PURPOSE

The purpose of this Administrative Control Directive is to specify the responsibilities and requirements for modifying the software on the process computer systems in order to control the integrity and to provide traceability of on-site software modification.

2.0 APPLICABILITY

This Administrative Control Directive is applicable to all personnel requesting or implementing software modifications to the process computer systems.

3.0 DEFINITIONS

3.1 Requestor - any person requesting a Process Computer software modification or identifying a problem with a Process Computer.

3.2 Designated Individual - the person designated as provided by ECD 5.21 to have immediate authority and responsibility for all software modifications for a given Software System.

3.3 Responsible Analyst - the person assigned to determine the specific software requirements for implementation of an approved software modification. This person has primary responsibility for implementation.

3.4 Process Computer - any real time computer system in operation within the nuclear department, the functional characteristics of which are determined by source code under direct control of Wisconsin Public Service.

3.5 Action Request - any program, data base, or system change/problem (see Form 1.9-1).

3.6 Program Change - any modification requiring the addition of a new program to the computer system or the modification of an existing system program, (see Form 1.9-2).

3.7 Data Base Change - any modification made to non-executable system data.

These changes may affect point definitions, log formats, or graphic displays, (see Form 1.9-3).

3.8 Software System - a group of related programs which perform similar or related functions for a particular process computer.

3.9 Master File - a file assigned by the System Software Technician to contain all signed original request forms and their attachments after completion of system changes.

3.10 System Maintenance Handbook - a compilation of guidelines, procedures, and checklists for each process computer that can be used while implementing system changes.

3.11 System Software Technician - the person designated by the Nuclear Computer Supervisor and whose responsibilities are herein defined.

4.0 RESPONSIBILITIES

4.1 Requestor Responsibilities

The Requestor shall:

4.1.1 Initiate action by completing and submitting a Process Computer Action Request (Form 1.9-1) to the System Software Technician for review.

4.1.2 Initiate a Process Computer Action Request (Form ACD 1.9-1) if a computer problem has occurred.

4.2 System Software Technician Responsibilities:

The System Software Technician shall:

4.2.1 Review all Process Computer Action Requests to determine if a change is required. If the change request was prepared based on a misunderstanding of the system operation, the System Software Technician will consult with the requestor to determine if the change request should be withdrawn.

4.2.2 Review all software change requests to determine 1) the affected Software System(s), and 2) the appropriate Designated Individual.

4.2.3 1) Determine whether the request requires a Program Change, Data Base Change, or other action.

2) Complete a Program Change Request Form (Form 1.9-2) and/or Data Base Change Request Form (Form 1.9-3) as determined in #1 above, and forward it along with the original request to the Designated Individual.

3) Issue Work Request if a hardware failure has occurred.

4.2.4 Determine if approved changes require changes to the simulation of the Honeywell Plant Process Computer. For changes which require changes to the simulator software, the System Software Technician will initiate a separate Process Computer Action Request Form as appropriate.

4.2.5 Ensure that all software changes are done in compliance with procedures, guidelines and checklists found in the Software Maintenance Handbooks.

4.2.6 Maintain and have responsibility for the Software Maintenance Handbooks.

4.2.7 Have primary responsibility for the scheduling and use of the computer system.

4.3 Designated Individual Responsibilities

The Designated Individual shall:

4.3.1 Review and approve software change requests.

4.3.2 Ensure that any additional control requirements such as second level reviews are performed prior to approving the request.

4.3.3 Complete the Software Safety Evaluation Report (Form 1.9-4) and attach to the initiating form. Engineering Control Direction 4.1, Design Change Control, provides guidance for the completion of the Report.

4.3.4 Forward copies of all completed forms ACD 1.9 to the Nuclear Computer Supervisor.

4.3.5 Initiate Design Change Requests (DCR) if required.

4.3.6 Act as administrator for all software changes which he approves, and ensure completion of all items delineated in this ACD and the Software Change Request Form(s).

4.3.7 Revise all plant procedures affected by the change.

4.4 Nuclear Computer Supervisor Responsibilities

The Nuclear Computer Supervisor shall:

4.4.1 Review changes approved by the Designated Individual to make a preliminary determination as to cost, scope, and manpower availability for the implementation of the change.

4.4.2 Determine if an approved software change requires hardware modification.

4.4.3 Assign the Responsible Analyst for the change.

4.5 Responsible Analyst Responsibilities

The Responsible Analyst shall:

4.5.1 Analyze the effect of data base changes on programs and other data base parameters in the computer system, in accordance with the data base change guidelines found in the Software Maintenance Handbooks.

4.5.2 Design, code, debug, test and integrate approved program changes assigned to him, in accordance with the programming conventions found in the Software Maintenance Handbook and in coordination with the System Software Technician.

4.5.3 Implement approved data base changes in accordance with the data base change guidelines found in the Software Maintenance Handbook, and in coordination with the System Software Technician. This applies to those data base changes found to be acceptable after adherance to paragraph 4.5.1.

4.5.4 Update all affected documentation and program listings in accordance with the Documentation Guidelines found in the Software Maintenance Handbook and in coordination with the System Software Technician.

4.5.5 Attach copies of all modified (updated) documentation to the Change Request Form(s).

4.5.6 At the completion of the software change, sign and forward the Process Computer Action Request and all associated forms to the master file and distribute copies per the distribution list found on the request form.

5.0 REQUIREMENTS

5.1 Procedural Requirements

5.1.1 Requests for system changes or the identification of computer system problems shall be made by submitting a completed Process Computer Action Request (Form 1.9-1) to the System Software Technician.

5.1.2 Requests for software changes as a result of simulator feedback reports or other simulator action will carry an additional designation which will allow cross-referencing of simulator documents to Software Change Status Reports.

5.1.3 Requests for program changes shall be made by submitting a completed Program Change Request Form (Form 1.9-2) to the appropriate Designated Individual.

5.1.4 Requests for data base changes shall be made by submitting a completed Data Base Change Request Form (Form 1.9-3) to the appropriate Designated Individual.

5.1.5 When requested program changes require associated data base changes, both request forms must be submitted together.

5.1.6 After determining that an approved program change assigned to him requires a data base change for which there is no accompanying signed request form, the Responsible Analyst shall note this on the Program Change Request Form and forward it to the Designated Individual for re-evaluation.

5.1.6 After determining that an approved software change assigned to him requires hardware modification not yet implemented, the Responsible Analyst is required to suspend implementation of the software change until a Design Change Request (DCR) has been approved for the hardware modification.

5.1.7 After determining that an approved software change assigned to him is not within his scope, the Responsible Analyst shall so note it on the Software Change Request Form and return it to the Designated Individual for re-evaluation.

5.1.8 The signed original Process Computer Action Request form(s) and all attachments shall be kept in the Master File after completion of the software change.

5.1.9 Copies of the signed original Process Computer Action Request form(s) and all attachments shall be distributed per the distribution list found on the request form.

5.2 Control Requirements

5.2.1 Programming conventions shall be printed in the Software Maintenance Handbook and must be followed when installing Program changes.

5.2.2 Data base change guidelines shall be printed in the Software Maintenance Handbook, and must be followed when making data base changes.

5.2.3 Documentation guidelines shall be printed in the Software Maintenance Handbook, and must be followed when making any software change.

5.2.4 The Designated Individual must approve all changes to software for which he has been assigned responsibility.

5.2.5 The Process Computer Program Change Request Form shall be signed by the Responsible Analyst to indicate that all requirements for software changes have been met.

5.2.6 The Designated Individual shall establish the testing requirements of any software changes approved, and shall establish the acceptance criteria.

5.2.7 The Responsible Analyst shall complete the Software Completion Form (ACD 1.9-5).

PROCESS COMPUTER ACTION REQUEST

DATE/TIME ______________________

SYSTEM: ______________________ REQUESTOR NAME: ______________

ACTION DESCRIPTION:

____

____

____

____

____

ACTION TAKEN:

____

____

____

____

____

ORIGINAL TO: System Software Technician

COPY TO: Assistant Manager - Plant Operations

RESOLUTION:

____

____

____

____

____

ASSOCIATED DOCUMENTS

NAME: ______________________

DATE: ______________________

PROCESS COMPUTER PROGRAM CHANGE REQUEST

Software Change No.__________________

A. Affected Software System(s) ______________________________

Designated Individual ______________________________

[ ] New Program   [ ] Existing Program Modification

Brief Statment of change:

[ ] Additional comments attached

____
Plant Computer Coordinator

Copies: Nuclear Computer Supervisor

Original: Designated Individual

B. [ ] Change Approved   [ ] Change Not Approved

Comments:

[ ] Additional comments attached

____
Designated Individual   Date

Copies: System Software Technician
Assistant Manager-Plant Operations

Original: Nuclear Computer Supervisor

C. Responsible Analyst assigned ______________________________

Comments:

____
Nuclear Computer Supervisor   Date

Copies: System Software Technician

Original: Responsible Analyst

Software Change No.______________________

At the completion of the change:

D. Effect on other existing programs or Data Base

[ ] No effect

Describe briefly, but adequately, the method by which this change was implemented. Attach copies of all modified program listings and other documentation which clearly illustrate and document this change.

Responsible Analyst Date

Copies: Nuclear Computer Supervisor
Assistant Manager-Plant Operations
Designated Individual

Original: System Software Technician

PROCESS COMPUTER DATA BASE CHANGE REQUEST

Software Change No.____________________

A. The requested Data Base Change requires (check all that apply):

[ ] Addition of new points

[ ] Deletion of existing points

[ ] Change attributes of existing points

The changes will affect (i.e., visibly alter):

[ ] Logs

[ ] Graphics

[ ] SAS

[ ] Other ____________________

Changes:

(Use back side if necessary)

____
System Software Technician Date

Designated Individual Assigned ____________________

Original: Designated Individual

B. Approvals:

____
Designated Individual Date

____
Nuclear Computer Supervisor Date

C. Changes Completed:

____
System Software Technician Date

Copies: Nuclear Computer Supervisor
Designated Individual
Original: System Software Technician

SOFTWARE SAFETY EVALUATION REPORT

Software Change Request No. ______________

1. Is the proposed change in conflict with Plant Technical Specifications?

   Yes ______ No ______

2. Is the probability of occurrence or consequences of an accident or malfunction of equipment important to safety previously evaluated in the USAR increased by this change or; is a possibility for an accident or malfunction of a different type than any evaluated previously in the USAR created by this change or; is the margin of safety as defined in the basis for any Technical Specification reduced by this change?

   Yes ______ No ______

   If 1 or 2 above is "Yes" prior NRC approval is required. If "No" continue with form.

3. Does proposed change constitute change in the facility or procedures as described in the USAR or conduct tests or experiments not described in the USAR?

   Yes ______ No ______

Delineate your reasoning below. Include enough detail to adequately illustrate that the change does not introduce an unreviewed safety question.

____

____

____

WPS SUPERVISOR/PERSON ______ DATE

Second level review of safety evaluation comments: ______________

____

____

Plant Technical Supervisor Date

SOFTWARE COMPLETION FORM

Software Change Number____________________

DCR (If Associated)____________________

Responsible Analyst____________________

Designated Individual____________________

Check and date each item as it is completed:

[ ]	Listing Generated Containing Changes	________
[ ]	System Design Documents (List):	________
	____________	________
	____________	________
	____________	________
[ ]	Test Procedure Performed	________
[ ]	Revised Test Procedure Filed	________

ATTACHMENT 5

SPDS LESSON PLAN

WISCONSIN PUBLIC SERVICE CORPORATION KEWAUNEE NUCLEAR POWER PLANT LESSON PLAN	NO. 770 TITLE: SAS for Operators, Day 2 PRESENTATION TIME:
	DATE: 3-06-85 — PAGE 1 of 19
REVIEWED BY: Lee Vanden Heuvel AUTHORED	APPROVED BY: T. P. Cenchlo

REFERENCES:

1. Technical Description of the Pressurized Water Reactor Safety Assessment System; Quadrex Corporation; January, 1982

2. Training Manual for Initial On-Site Safety Assessment System (SAS) Training Program, Quadrex Corporation; January, 1982

AUDIO VISUALS: Slides, overheads

TEACHING STRATEGIES OR METHODS: Lecture

EVALUATION METHODS
OF TRAINEES' PROGRESS:

JOB-RELATED TASKS:

PERFORMANCE OBJECTIVES

TERMINAL OBJECTIVES:

ENABLING OBJECTIVES:

1. RECOGNIZE the purpose of the SAS.

2. EXPLAIN the organization of the SAS subsystems including:

   a. the name of each subsystem
   b. where each subsystem is displayed
   c. explain how the primary SAS display indicates an off-normal condition on the:

      1. Safety System Readiness Monitor (SSRM)
      2. Safety System Performance Monitor (SSPM)
      3. Critical Safety Function Monitor (CSFM)

3. RECOGNIZE the parameters displayed on the 3 top level displays.

4. STATE the purpose of the Accident Identification and Display System (AIDS) and RECOGNIZE the four events it monitors.

5. RECOGNIZE the parameters contained in each Trend Graph Group.

6. EXPLAIN how the value displayed on the SAS relates to other control room indications for the following parameters:

   a. RCS Pressure
   b. PRZR Level
   c. RCS Cold Leg Temperature
   d. RCS Hot Leg Temperature
   e. Charging - Letdown Flow
   f. Auxiliary Feedwater Flow
   g. Steam Generator Level
   h. Steam Generator Pressure
   i. Secondary Radiation Target
   j. Containment Environment Target
   k. Containment Radiation
   l. Reactor Vessel Level
   m. Subcooling
   n. Core Exit Temperature
   o. Power Level
   p. RCS Average Temperature
   q. Nuclear Instrumentation
   r. RHR Heat Exchanger inlet and outlet temperature
   s. RHR Flow
   t. Containment Pressure
   u. Steam Generator Flows (Steam and Feed)
   v. Stack Effluent Radiation

7. EXPLAIN how SAS uses a running average to determine some of its setpoints and how a setpoint determined by this method differs from a fixed setpoint.

8. EXPLAIN how data displayed on the SAS is validated and how the results of this validation are displayed.

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING | TEACHING AIDS/ INSTRUCTOR NOTES

I. INTRODUCTION

A. Introduce yourself

B. Explain rules for breaks

1. 10 minutes every hour

C. Overview of class

1. Review material from last time first

2. Most of this lecture on inputs to SAS, setpoints and validation

II. REVIEW

A. Review Objectives — Transp. #1, Handout #1

1. Explain objectives

a) 1. "Recognize the purpose of the SAS"

b) 2. "Explain the organization of the SAS subsystems"

(1) Name the subsystems

(2) Where each is displayed

(3) Connection between the primary and secondary subsystems. How the primary indicates off-normal conditions of the secondary.

c) 3. "Recognize the parameters on the 3 top level displays"

(1) NORMAL

(2) HEATUP/COOLDOWN

(3) COLD SHUTDOWN

d) 4. "State the purpose of the AIDS and recognize the four events it monitors"

(1) Since it is not implemented in the Control Room, no details, just the purpose and the events if monitors.

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING | TEACHING AIDS/ INSTRUCTOR NOTES

e) 5. "Recognize the parameters contained in each trend group."

(1) Given what it says on the Keyboard, recognize the parameters in that group.

f) 6. EXPLAIN how the value displayed on the SAS relates to other control room indications for the following parameters:

a. RCS Pressure
b. PRZR Level
c. RCS Cold Leg Temperature
d. RCS Hot Leg Temperature
e. Charging - Letdown
f. Auxiliary Feedwater Flow
g. Steam Generator Level
h. Steam Generator Pressure
i. Secondary Radiation Target
j. Containment Environment Target
k. Containment Radiation
l. Reactor Vessel Level
m. Subcooling
n. Core Exit Temperature
o. Power Level
p. RCS Average Temperature
q. Nuclear Instrumentation
r. RHR Heat Exchanger inlet and outlet temperature
s. RHR Flow
t. Containment Pressure
u. Steam Generator Flows (Steam and Feed)
v. Stack Effluent Radiation

(1) Most of these are easy because they are self explanitory and the same as the instrumentation you normally use, but some of them are not obvious or are not the instrumentation you normally use. We will look at those in more detail.

g) 7. "EXPLAIN how SAS uses a running average to determine some of its setpoints and how a setpoint determined by this method differs from a fixed setpoint."

(1) For some parameters, the setpoint changes according to the parameter history: i.è., containment temperature, 110°F in winter may be an alarm, 110°F in summer it may not be

h) 8. "EXPLAIN how data displayed on the SAS is validated and how the results of this validation are displayed.

(1) Know the different ways a parameter is validated and the way the validation results are represented on the screen.

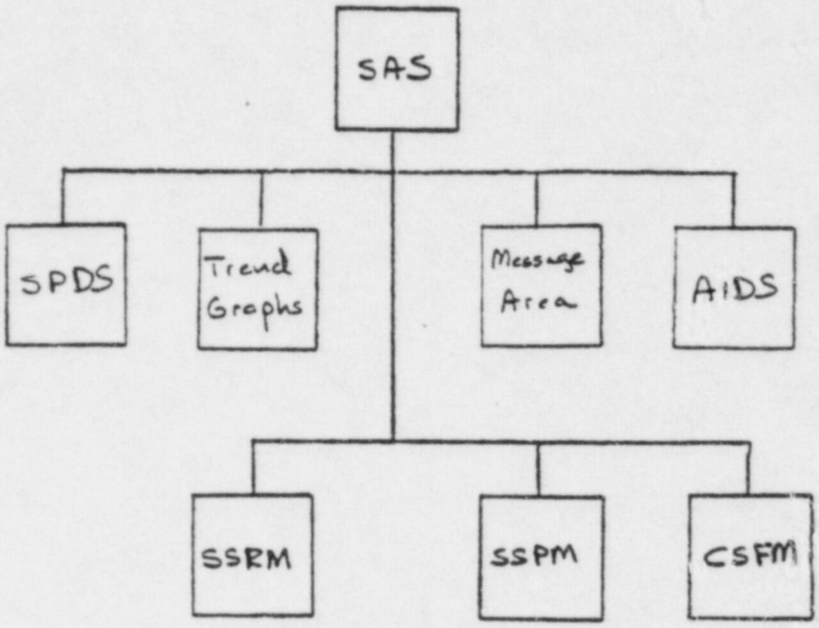
2. Objectives are testable

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING	TEACHING AIDS/ INSTRUCTOR NOTES
B. Review the material for the objectives covered last time	
1. Purpose of SAS	Transp. #2 Handout #2 Objective 1
a) Provide continous indication of parameters representative of the safety status of the plant	
b) Aid the operator in the rapid detection of abnormal conditions	
c) Concentrate in one location a minimum set of parameters to allow timely status assessment	
d) Incorporate human factors considerations	
e) Indentify faulty data	
f) Display information under all plant conditions	
(1) Under all plant conditions, allow the operator to quickly assess the safety status of the plant by looking at only one display.	

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING | TEACHING AIDS/ INSTRUCTOR NOTES

2. Organization of Subsystems

Transp. #3
Handout #3
Objective 2



LESSON PLAN OUTLINE FOR CLASSROOM TRAINING | TEACHING AIDS/ INSTRUCTOR NOTES

a) Primary Subsystems

(1) Displayed on SAS Terminal

(2) Accessed by SAS keyboard

(3) Primary Subsystems

(a) SPDS (Safety Parameter Display System) - parameters used to assess plant safety status

(b) Trend Graphs - 30 minute trends of related sets of parameters

(c) Message Area - Indicates MODE, DATE, TIME, T_{avg}, Power; Event Markers; Also indicates off normal status of secondary subsystems (when secondary subsystems implemented)

(d) AIDS (Accident Indentification and Display System) Graphically depicts the liklihood that the following 4 events may be occuring: — Objective 4

i) LOCA - Loss of Coolant Accident

ii) SGTR - Steam Generator Tube Rupture

iii) LOSC - Loss of Secondary Coolant

iv) ICC - Inadequate Core Cooling

b) Secondary Subsystems

(1) Displayed on Honeywell Terminal

(2) Accessed by Honeywell Keyboard

(3) Line in message area on SAS screen indicated off normal condition of these systems

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING | TEACHING AIDS/ INSTRUCTOR NOTES

(4) Secondary Subsystems

(a) SSRM (Safety System Readiness Monitor) - Determines the availability of selected safety systems

(b) SSPM (Safety System Performance Monitor) - Assesses safety system performance

(c) CSFM (Critical Safety Function Monitor) - Assesses status of CSF Trees

These were discussed last time. They are not implemented now, and will not be unless operations asks for them. The problem is lack of inputs into the Honeywell. I wanted to clarify the capabilities of these subsystems if implemented.

SSRM - Readiness Monitor

Monitors six systems for readiness:

1. Safety Injection
2. Diesel Generators
3. 480V Essential Power
4. 125V DC Power
5. Service Water
6. Component Cooling Water

Can list the components of each system that is off-normal.

Also can draw P&IDs showing the operation of the system and the off-normal condition.

SSPM - Performance Monitor

Has intelligence. Points out things that may not be obvious immediately.

CONTAINMENT SPRAY -
Containment pressure >23 psig and combined ICS flow <1200 then "Inadequate Containment Spray Flow".

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING | TEACHING AIDS/ INSTRUCTOR NOTES

AFW
SGNRL <13 and flow <50 "Inadequate AFW to A SG".

CONTAINMENT ISOLATION
If containment isolation called for and all valves don't close "11 containment isolation valves mispositioned".

SERVICE WATER
SI and not SWP1 "Service water pump 1A1 failed to start".

SI Checks

(a) "One train SI failed to initiate"

(b) By looking at RCS pressure and SI pump curves can print "Inadequate SI flow for existing pressure".

(c) "Cold leg injection blocked"

(d) "Vessel injection blocked"

(e) "Safety Injection Termination Criteria NOT met"

<u>CSFM -</u>

Monitors status of trees automatically. Graphically shows the branches you're on. Indicates inserted points and bad input.

3. Top Level Displays

Objective 3

a) Should indicate safety status for that mode

b) 3 modes to cover all modes of plant operation

c) Normal Top Level Display

Slide #1
Handout #4

(1) Point out 3 major areas

(a) SPDS

(b) Message Area

(c) AIDS

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING — TEACHING AIDS/INSTRUCTOR NOTES

(2) Review parameters in each major area

(a) SPDS

i) RCS Pressure

ii) PRZR Level

iii) Hot Leg Temperature

iv) Cold Leg Temperature

v) Charging - Letdown

vi) Aux Feed Flow

vii) S/G Level

ix) Secondary Radiation

(1) Air ejector radiation

(2) S/G Blowdown radiation

x) Containment Environment

(1) Temperature — *Only containment pressure is currently input.*

(2) Pressure

(3) Humidity

(4) Sump Level

xi) Containment Radiation

xii) Reactor Vessel Level

xiii) Subcooling

(1) Subcooled

(2) Saturated

(3) Superheated

xiii) Core Exit Temperature

(b) Message Area

i) Date

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING	TEACHING AIDS/ INSTRUCTOR NOTES
ii) Time	
iii) Mode of SAS	
iv) Power	
v) RCS Avg Temperature	
vi) Event markers	
vii) Secondary system messages	
(c) AIDS	
i) AIDS bars	
ii) AIDS Targets	
iii) Not displayed in CR until validated on simulator	
d) Heatup/Cooldown Top Level Display.	
(1) Same as Normal Top Level Display except for:	Slide #2 Handout #5
(a) Limits change as plant conditions change for:	
i) RCS Pressure	
ii) Cold Leg Temperature	
(b) Limits disappear for:	
i) PRZR Level	
ii) S/G Level	
iii) S/G Pressure	
(c) Message Area	
i) Intermediate Start-up Rate	
e) Cold Shutdown Top Level Display	Slide #3 Handout #6
(1) Slower response time	
(2) 3 groups of trends with numerical values, bar graphs and 2 hour trend graphs for:	

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING — TEACHING AIDS/ INSTRUCTOR NOTES

(a) Source Range A and B

(b) RHR

i) Flow

ii) HX inlet temperature

iii) HX outlet temperature

(c) RCS parameters

i) RCS pressure

ii) Core Exit Temperature

iii) Vessel Level

f) Limits on parameters — Slide #4

(1) Bar graphs

(a) Bar changes color

(2) Targets

(a) Target changes color

(3) Numerical Values

(a) Red box appears around number

g) Event markers — Handout #7

(1) Shown for

(a) Reactor Trip

(b) Link Down

(c) SI actuation

(d) Feedwater actuation

i) These event numbers also show up on the trend graphs

h) Secondary Subsystem off-normal markers

(1) SSRM - Status Off Normal

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING	TEACHING AIDS/ INSTRUCTOR NOTES
(2) SSPM - Status Off Normal	
(3) CSFM - Status Off Normal	
4. Trend Graphs	Slide #5
a) 30 minute time histories	
b) bar graphs with the same color as the trend	
c) 1st trend group appears in lower right	
d) 2nd trend group again apprear in lower right with the first group moving up to the upper right	Slide #6
e) Event markers appear on trend graphs and scroll across display	
f) Point out difference between wide and narrow range T hot and T cold:	
(1) Wide range just an expanded scale. It comes off the same sensors, (Wide range RTDs)	
g) RCS Pressure versus Temperature Curve	Slide #7
(1) Dot, with a 1 hour snake behind it	
h) Review trend graph groupings	Transp. #4, #5 Handout #8, #9
(1) Point out unusual groupings	Objective 6
(a) CONTAIN HHT - none of these are input	
(b) RCS - Pressure, PRZR Level, Core exit temperature, subcooling	
(c) Core Cooling - Rx Vessel Level, core exit temperature, subcooling	
(d) Tank Levels - sump, RWST, CST, none of these are input	
(e) Rad Monitors - Containment Radiation (R - 2) Air ejector, SGBD, stack effluent (R - 14)	

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING	TEACHING AIDS/ INSTRUCTOR NOTES
5. Keyboards	Transp. #6 Handout #10

a) Primary keyboard

(1) Mode Keys

(a) Selects the top level display

(b) Complete screen affected

(2) AIDS Keys

(a) AIDS bars and message area remain as is

(b) SPDS parameters disappear and AIDS parameter appear

(3) Trend Graphs

(a) AIDS bars and message area remain as is

(b) First trend graph appears in lower right, second also appears in lower right and first moves to upper right

b) Secondary Keyboard

(1) Won't go over this until the secondary subsystems are implemented.

III. Validation of Input

A. We will review the inputs for each of the parameters that SAS displays. However, I would first like to briefly cover how SAS validates the inputs to it.

B. Validation - an attempt to check incoming data to determine whether it is meaningful or not.

C. When the system is fully implemented all parameters will have this validation scheme. Presently only some have it.

D. Look at inputs with different numbers of channels.

1. Single channel input

a) Example - Aux. Feed Flow

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING | TEACHING AIDS/ INSTRUCTOR NOTES

b) If it is out of range, then the number replaced by the word FAIL in yellow. (0 - 450)

2. Two channel input

a) Example - Steam flow

b) Does a range check. If it throws one out then the number displayed in yellow. If it throws both out then the word FAIL appears in yellow. (0 - 4,000,000)

c) Does a spread check. If the spread too large, then the value is displayed in yellow.

3. Three channel input

a) Example - S/G Pressure

b) Does a range check. (0-1400)
Throws one out ===> nothing
Throws out two ===> parameter value in yellow
Throws them all out ===> FAIL in yellow

c) Also checks the values against each other. Examples:

(1) With inputs of 850
840
425
SAS throws out the 425 because the 840 and 850 validate each other. Therefore, the displayed result is 845.

4. More than three channel input

a) Example - Incore Thermocouples

b) SAS performs this same type of comparison with more inputs. It compares the value of each input to all the other inputs. If the value seems way off compared to the others, SAS ignores that value.

c) This is how SAS gets Core Exit Temperature right away. It automatically ignores those values that aren't close to the others.

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING	TEACHING AIDS/ INSTRUCTOR NOTES

E. Summary of Displaying Validation Results

Transp. #7
Handout #11

DISPLAY	CONDITION
FAIL in yellow	No sensors left within range
Parameter in yellow (Alert)	a. Only 1 sensor left in range (2 or more to start with) b. Two sensors left, large spread
Parameter in white (O.K.)	a. 1 of 1 sensors in range b. 2 sensors in range, narrow spread c. More than 2 sensors

FAIL - No inputs left
Alert - Displays the result, but it may not be accurate
O.K. - Parameter is acceptable

1. Remember the change of a number to RED has to do with its setpoint, not with its validation.
   YELLOW indicates caution in using the parameter value displayed.
   RED indicates the value is in alarmed condition.

   At present there is a problem since not all have the SAS validation scheme.

   A non-valid parameter in alarm appears white, just like a valid parameter in alarm. SAS can only pass one of the following:

   Alarm
   Non-Valid
   O.K.

   With the precidence in the order listed. Therefore once alarmed, the validation information is gone.

IV. Parameter Inputs

(Refer to the transparencies)

Transp. #8 - 13
Handout #12 - 17

A. Additional Notes

LESSON PLAN OUTLINE FOR CLASSROOM TRAINING — TEACHING AIDS/ INSTRUCTOR NOTES

1. RCS Pressure

   a) Explain how RCS pressure limits are determined by RCS Average Temperature for heatup/ cooldown mode. — Slide #1, 2; Handout #4, 5

   b) RCS cold leg temperatures determined by RCS pressure for heatup/cooldown mode — Transp. #14; Handout #18

2. Core Exit Temperature

   a) Example of SAS validation of a parameter with more than 3 inputs.

3. Containment Environment is a containment pressure alarm since only that parameter is input.

V. Miscellaneous Information

A. When link down occurs

   1. ~~Data is as is~~ No data displayed, trends still available.

   2. Brown turns to yellow

B. Stuck key

   1. STUCK KEY message appears when a key is stuck. Clears when the stack of input keystrokes clears.

VI. REVIEW

A. Top Level Displays

   1. Normal — Slide #1

      a) SPDS area

      b) Message area

      c) AIDS area

   2. Heatup/cooldown — Slide #2

      a) Save as NORMAL except for the limits and startup rate

   3. Cold shutdown — Slide #3

SAFETY ASSESSMENT SYSTEM (SAS) TRAINING

OBJECTIVES

1. RECOGNIZE THE PURPOSE OF THE SAS.
2. EXPLAIN THE ORGANIZATION OF THE SAS SUBSYSTEMS INCLUDING:
   - A. THE NAME OF EACH SUBSYSTEM
   - B. WHERE EACH SUBSYSTEM IS DISPLAYED
   - C. EXPLAIN HOW THE PRIMARY SAS DISPLAY INDICATES AN OFF-NORMAL CONDITION ON THE:
     1. SAFETY SYSTEM READINESS MONITOR (SSRM)
     2. SAFETY SYSTEM PERFORMANCE MONITOR (SSPM)
     3. CRITICAL SAFETY FUNCTION MONITOR (CSFM)
3. RECOGNIZE THE PARAMETERS DISPLAYED ON THE 3 TOP LEVEL DISPLAYS.
4. STATE THE PURPOSE OF THE ACCIDENT IDENTIFICATION AND DISPLAY SYSTEM (AIDS) AND RECOGNIZE THE FOUR EVENTS IT MONITORS.
5. RECOGNIZE THE PARAMETERS CONTAINED IN EACH TREND GRAPH GROUP.
6. EXPLAIN HOW THE VALUE DISPLAYED ON THE SAS RELATES TO OTHER CONTROL ROOM INDICATIONS FOR THE FOLLOWING PARAMETERS:
   - A. RCS PRESSURE
   - B. PRZR LEVEL
   - C. RCS COLD LEG TEMPERATURE
   - D. RCS HOT LEG TEMPERATURE
   - E. CHARGING - LETDOWN
   - F. AUXILIARY FEEDWATER FLOW
   - G. STEAM GENERATOR LEVEL
   - H. STEAM GENERATOR PRESSURE
   - I. SECONDARY RADIATION TARGET
   - J. CONTAINMENT ENVIRONMENT TARGET
   - K. CONTAINMENT RADIATION
   - L. REACTOR VESSEL LEVEL
   - M. SUBCOOLING
   - N. CORE EXIT TEMPERATURE
   - O. POWER LEVEL
   - P. RCS AVERAGE TEMPERATURE
   - Q. NUCLEAR INSTRUMENTATION
   - R. RHR HEAT EXCHANGER INLET AND OUTLET TEMPERATURE
   - S. RHR FLOW
   - T. CONTAINMENT PRESSURE
   - U. STEAM GENERATOR FLOWS (STEAM AND FEED)
   - V. STACK EFFLUENT RADIATION
7. EXPLAIN HOW SAS USES A RUNNING AVERAGE TO DETERMINE SOME OF ITS SETPOINTS AND HOW A SETPOINT DETERMINED BY THIS METHOD DIFFERS FROM A FIXED SETPOINT.
8. EXPLAIN HOW DATA DISPLAYED ON THE SAS IS VALIDATED AND HOW THE RESULTS OF THIS VALIDATION ARE DISPLAYED.

PURPOSE OF SAS

1) PROVIDE A CONTINUOUS INDICATION OF PLANT PARAMETERS OR DERIVED VARIABLES, REPRESENTATIVE OF THE SAFETY STATUS OF THE PLANT

2) AID THE OPERATOR IN THE RAPID DETECTION OF ABNORMAL OPERATING CONDITIONS

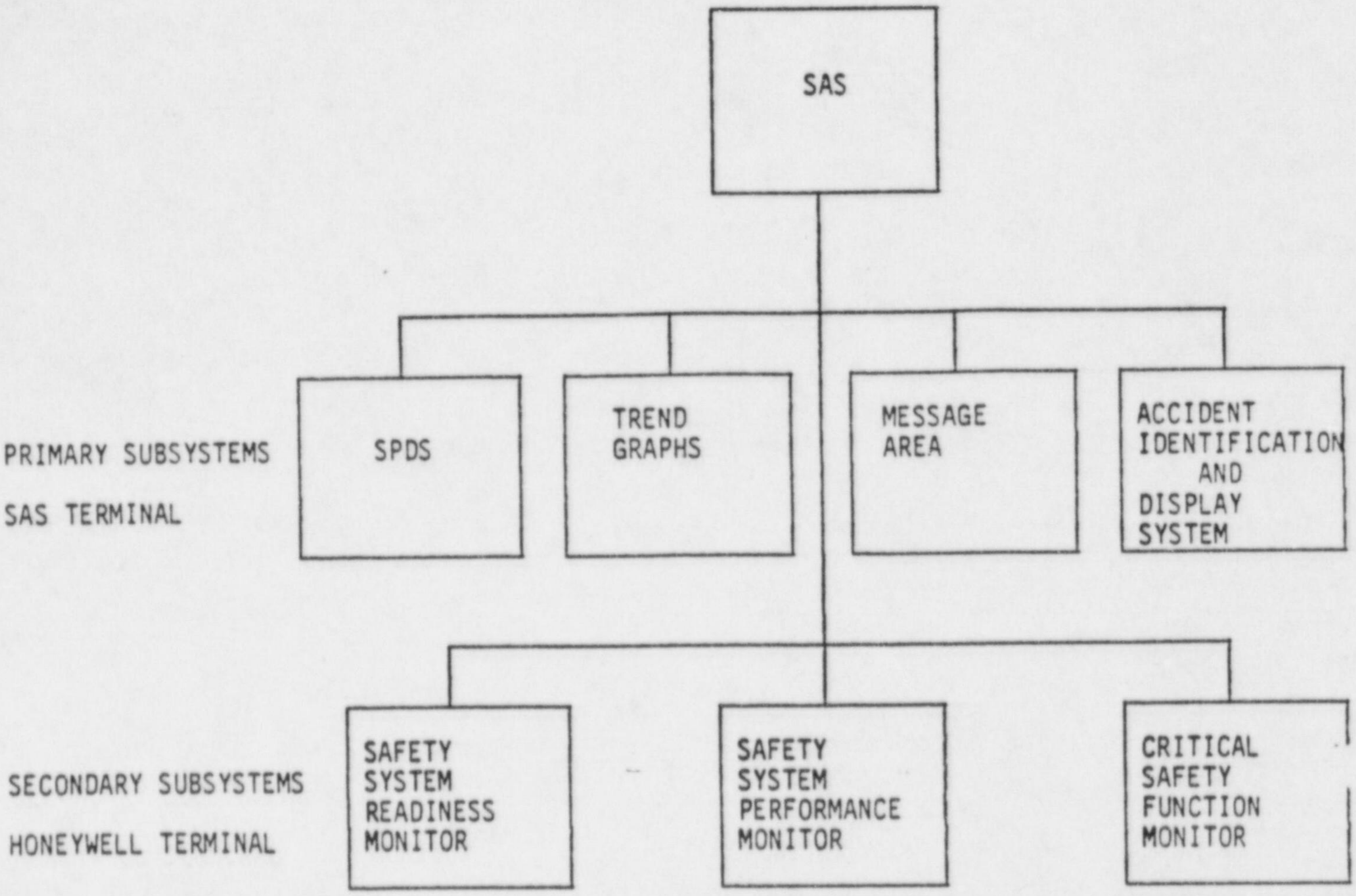
3) CONCENTRATE IN ONE LOCATION A MINIMUM SET OF PARAMETERS TO ALLOW TIMELY STATUS ASSESSMENT WITHOUT SURVEYING THE ENTIRE CONTROL ROOM

4) INCORPORATE HUMAN FACTORS CONSIDERATIONS

5) IDENTIFY FAULTY DATA

6) DISPLAY INFORMATION DURING STEADY-STATE AND TRANSIENT CONDITIONS

ORGANIZATION OF SAS SUBSYSTEMS



Primary subsystems

SPDS - (Safety Parameter Display System) - parameters used to assess plant safety status

Trend Graphs - 30 minute trends of related sets of parameters.

Message Area - Indicates MODE, DATE, TIME, Tavg, POWER; event markers; also indicates off-normal status of secondary subsystems (only when secondary subsystems implemented).

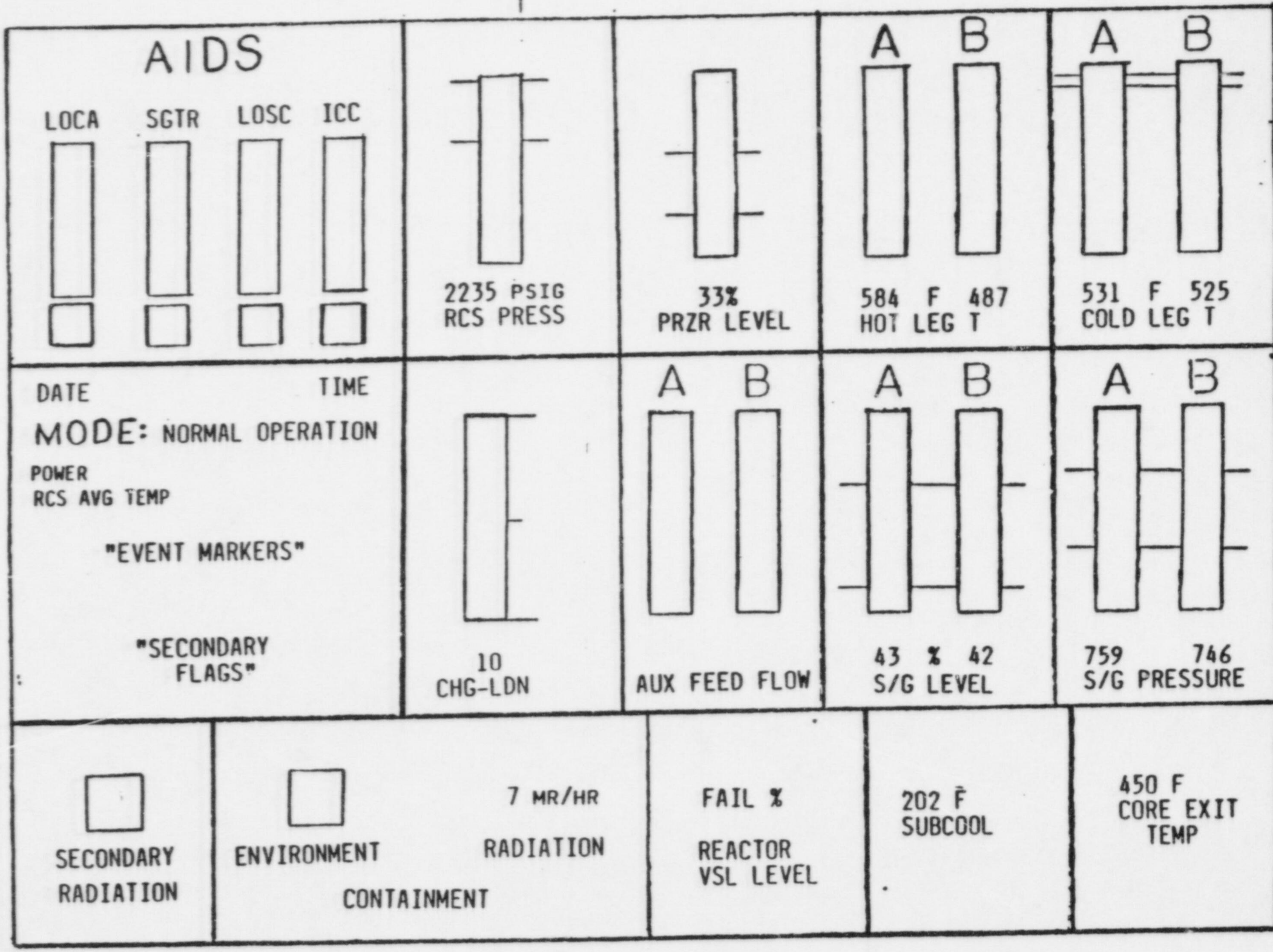
AIDS - (Accident Identification and Display System) - graphically depicts the likelihood that the following 4 events may be occuring:

1) LOCA - Loss of Coolant Accident

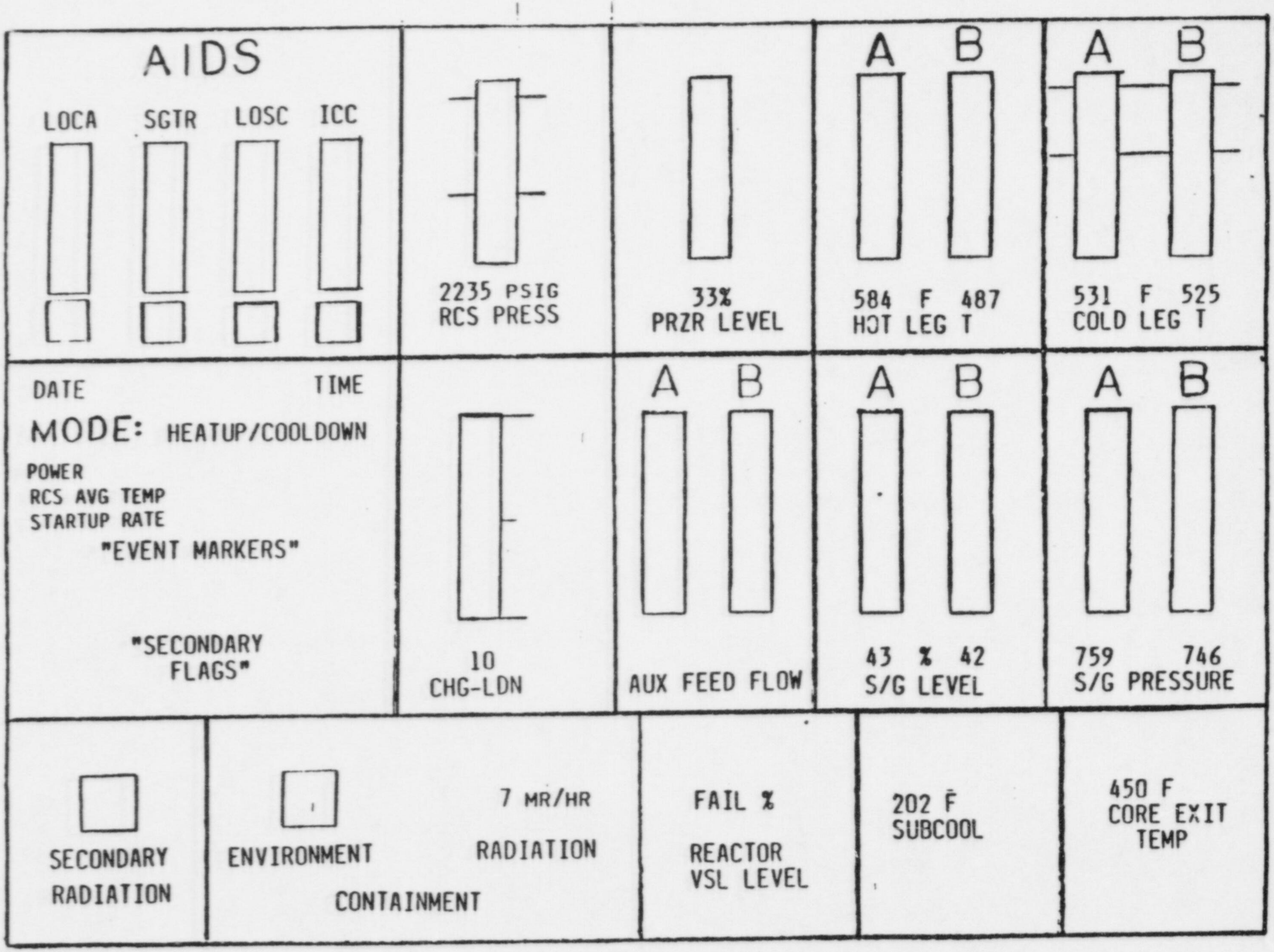
2) SGTR - Steam Generator Tube Rupture

3) LOSC - Loss of Secondary Coolant (Steam and Feed Breaks)

4) ICC - Inadequate Core Cooling



AIDS
LOCA
SGTR
LOSC
ICC
2235 PSIG
RCS PRESS
33%
PRZR LEVEL
A
B
584 F 487
HOT LEG T
A
B
531 F 525
COLD LEG T
DATE
TIME
MODE: NORMAL OPERATION
POWER
RCS AVG TEMP
"EVENT MARKERS"
"SECONDARY FLAGS"
10
CHG-LDN
A
B
AUX FEED FLOW
A
B
43 % 42
S/G LEVEL
A
B
759 746
S/G PRESSURE
SECONDARY RADIATION
ENVIRONMENT
7 MR/HR
RADIATION
CONTAINMENT
FAIL %
REACTOR VSL LEVEL
202 F
SUBCOOL
450 F
CORE EXIT TEMP


AIDS
LOCA
SGTR
LOSC
ICC
2235 PSIG
RCS PRESS
33%
PRZR LEVEL
A
B
584 F 487
HOT LEG T
A
B
531 F 525
COLD LEG T
DATE
TIME
MODE: HEATUP/COOLDOWN
POWER
RCS AVG TEMP
STARTUP RATE
"EVENT MARKERS"
"SECONDARY FLAGS"
10
CHG-LDN
A
B
AUX FEED FLOW
A
B
43 % 42
S/G LEVEL
A
B
759 746
S/G PRESSURE
SECONDARY RADIATION
ENVIRONMENT
7 MR/HR
RADIATION
CONTAINMENT
FAIL %
REACTOR VSL LEVEL
202 F
SUBCOOL
450 F
CORE EXIT TEMP

	SOURCE RANGE A	SOURCE RANGE B	2 HOUR SOURCE RANGE TREND
RHR FLOW	RHR Hx OUTLET TEMP	RHR Hx INLET TEMP	2 HOUR RHR TREND
RCS PRESS	CORE EXIT TEMP	Rx VESSEL LEVEL	2 HOUR RCS GROUP TREND
DATE		TIME	MODE: COLD SHUTDOWN

MESSAGE AREA

LINE #	MESSAGE	UNITS/ACT. TIME	NOTES
1	DATE, TIME		1
2	MODE: NORMAL OR, HEATUP/COOLDOWN		1
3	UNIT NAME		1
4	POWER LEVEL	% OR, AMPS OR, CPS	1
5	RCS AVG TEMP	F	1
6	STARTUP RATE	DPM	2
7	1 REACTOR TRIP	ACT. TIME	3
8	2 LINK DOWN	ACT. TIME	3
9	3 S. I. ACTUATION	ACT. TIME	3
10	4 FEEDWATER ISOLATION	ACT. TIME	3
11	SSRM - STATUS OFF NORMAL		3, 4
12	SSPM - STATUS OFF NORMAL		3, 4
13	CSF - STATUS OFF NORMAL		3, 4

NOTES:

1. ALWAYS DISPLAYED DURING NORMAL AND HEATUP/COOLDOWN MODES.
2. DISPLAYED DURING HEATUP/COOLDOWN MODE ONLY.
3. DISPLAYED ONLY WHEN AN OFF-NORMAL CONDITION EXISTS.
4. NOT DISPLAYED UNTIL SECONDARY SUBSYSTEMS IMPLEMENTED.

TREND GRAPH GROUPINGS

GROUP NAME LABEL	CONTENTS (TREND PARAMETERS)
NUCLEAR INSTR.	POWER RANGE POWER INTERMEDIATE RANGE POWER SOURCE RANGE POWER
CONTAIN LPR	* CONTAINMENT SUMP LEVEL CONTAINMENT PRESSURE CONTAINMENT RADIATION
CONTAIN HHT	* CONTAINMENT HYDROGEN CONCENTRATION * CONTAINMENT HUMIDITY * CONTAINMENT TEMPERATURE
RCS	REACTOR COOLANT SYSTEM (RCS) PRESSURE PRESSURIZER LEVEL CORE EXIT TEMPERATURE SUBCOOLING
RCS TEMP WR	RCS LOOP A HOT LEG TEMPERATURES RCS LOOP A COLD LEG TEMPERATURES RCS LOOP B HOT LEG TEMPERATURES RCS LOOP B COLD LEG TEMPERATURES
RCS TEMP NR (NOTE: NR IS JUST THE WR WITH AN EXPANDED SCALE FOR EASIER READING)	RCS LOOP A HOT LEG TEMPERATURES RCS LOOP A COLD LEG TEMPERATURES RCS LOOP B HOT LEG TEMPERATURES RCS LOOP B COLD LEG TEMPERATURES
CORE COOLING	* REACTOR VESSEL LEVEL CORE EXIT TEMPERATURE SUBCOOLING
S/G PRESS	STEAM GENERATOR A PRESSURE STEAM GENERATOR B PRESSURE
S/G LEVELS	STEAM GENERATOR A LEVEL STEAM GENERATOR B LEVEL
S/G FLOWS	STEAM GENERATOR A-FF STEAM GENERATOR A-SF STEAM GENERATOR B-FF STEAM GENERATOR B-SF

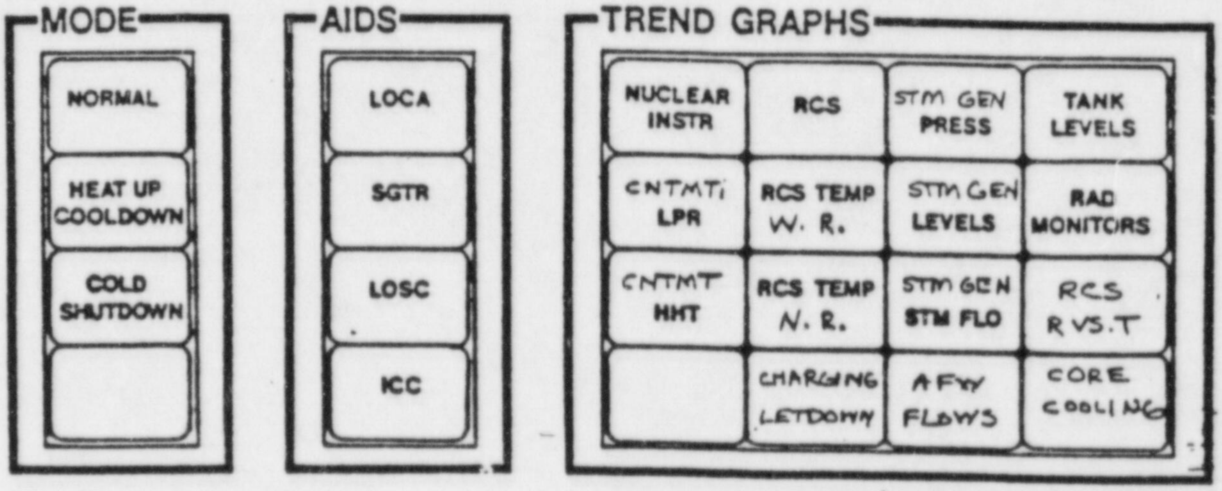
TREND GRAPH GROUPINGS

GROUP NAME LABEL	CONTENTS (TREND PARAMETERS)
AFW FLOWS	* AUXILIARY FEEDWATER FLOW TO S/G A * AUXILIARY FEEDWATER FLOW TO S/G B
TANK LEVELS	* CONTAINMENT SUMP LEVEL * REFUELING WATER STORAGE TANK * CONDENSATE STORAGE TANK
RADIATION MONITORS	CONTAINMENT RADIATION AIR EJECTOR RADIATION STEAM GENERATOR BLOWDOWN RADIATION STACK EFFLUENT RADIATION
RCS P VS. T	RCS PRESSURE VS. RCS AVERAGE TEMPERATURE
CHARGING LETDOWN	CHARGING FLOW, LETDOWN FLOW

* NOT INPUT TO SAS - VALUE DISPLAYED AS "FAIL"

SAS-OP-2
Transp. #6
Handout #10

PRIMARY CRT FUNCTION KEYBOARD



DISPLAY	CONDITION
FAIL IN YELLOW	NO SENSORS IN RANGE
PARAMETER IN YELLOW (ALERT)	A. ONLY 1 SENSOR LEFT IN RANGE (2 OR MORE TO START WITH) B. TWO SENSORS LEFT, LARGE SPREAD BETWEEN THEM
PARAMETER IN WHITE (O.K.)	A. 1 OF 1 SENSORS IN RANGE B. 2 SENSORS IN RANGE, NARROW SPREAD C. MORE THAN 2 SENSORS

FAIL - NO INPUTS LEFT
ALERT - DISPLAYS THE RESULT, BUT IT MAY NOT BE ACCURATE
O.K. - PARAMETER IS ACCEPTABLE

Parameter	Inputs	Comments/Limits
RCS Pressure	PRZR Pressure (NR) P429 P430 P431 P449 RCS Pressure (WR) P420	Uses average of PRZR press. channels when this average is >1700. If average of PRZR press. channels is <1700, then the WR is used. Limits for normal mode are 1816-2484 psig Limits for heatup/cooldown mode is determined by RCS Average Temperature and RCS Pressure vs. Temperature Curve (see diagram page 18)
Pressurizer Level	PRZR Level L426 L427 L428	Average of these three Limits for normal mode are 18-56% No limit for heatup/cooldown
Hot Leg Temperatures	Hot Leg WR RTDs A - T450A B - T451A	No limits

PARAMETER	INPUTS	COMMENTS/LIMITS
COLD LEG TEMPERATURES	COLD LEG WR RTDs A - 450B B - 451B	LIMIT FOR NORMAL MODE ARE 521-553°F LIMIT FOR HEATUP/COOLDOWN MODE DETERMINED BY THE RCS AVERAGE PRESSURE AND RCS PRESSURE VS. TEMPERATURE CURVE (SEE DIAGRAM PAGE 18)
CHARGING LETDOWN	CHARGING FLOW F128 LETDOWN FLOW F134	
AUX FEED FLOW	NONE AT PRESENT	
STEAM GENERATOR LEVEL	NARROW RANGE LEVELS S/G A L461 L462 L463 S/G B L471 L472 L473	LIMITS FOR NORMAL MODE ARE 16-65% NARROW RANGE AVERAGE (S/G A) NARROW RANGE AVERAGE (S/G B)

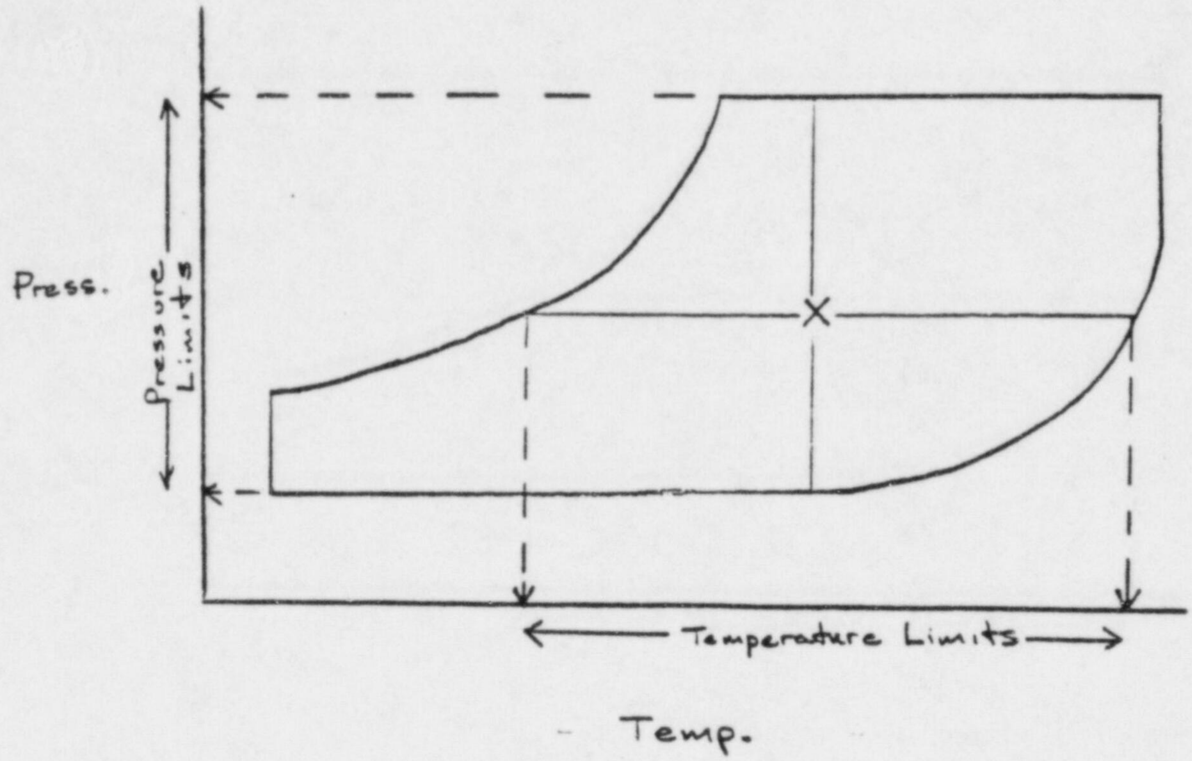
PARAMETER	INPUTS	COMMENTS/LIMITS
STEAM GENERATOR PRESSURE	A P468 P469 P482 B P478 P479 P483	LIMITS FOR NORMAL MODE ARE 500-1050 PSIG
SECONDARY RADIATION	R-15 R-19	ALARM AT >1000 CPM ALARM AT >6000 CPM
CONTAINMENT Radiation	R-2	ALARMS AT TWICE THE RUNNING AVERAGE, RESET EVERY 15 MINUTES (NOTE: SENSITIVITY DECREASES AS LEVEL INCREASES)
REACTOR VESSEL LEVEL	RVLIS - N/A STATE VESSEL LEVEL DURING REFUELING	

PARAMETER	INPUTS	COMMENTS/LIMITS
SUBCOOLING	RCS PRESS - SEE PREVIOUS DEFINITION CORE EXIT TEMPERATURE - SEE BELOW	1. BASED ON RCS PRESSURE AND CORE EXIT TEMPERATURE 2. ±1°F DEADBAND 3. TEXT CHANGES FROM SUBCOOLED - WHITE SATURATED - YELLOW SUPERHEAT - RED
CORE EXIT TEMPERATURE	INCORE THERMOCOUPLES	SHOULD BE CLOSE TO HONEYWELL AVERAGE. SAS VALIDATION SCHEME.
OTHER PARAMETERS		
NUCLEAR INSTRUMENTATION SOURCE RANGE INTERMEDIATE RANGE POWER RANGE	 SR31 SR32 IR35 IR36 PR41 PR42 PR43 PR44	MESSAGE AREA POWER DISPLAYED AS: A) CPM IF ((POWER RANGE <1% OR TRIP BREAKERS OPEN) AND SR ENERGIZED) B) AMPS IF ((POWER RANGE >1% OR TRIP BREAKERS OPEN) AND SR <u>NOT</u> ENERGIZED) C) % POWER IF TRIP BREAKERS CLOSED AND PR >1% (NOTE: DURING ATWS WOULD DISPLAY IN AMPS IF TRIP BREAKERS OPEN)

PARAMETER	INPUTS	COMMENTS/LIMITS
SG A FEEDFLOW	F466 F467	
SG A STEAM FLOW	F464 F465	
SG B FEEDFLOW	F474 F475	
SG B STEAM FLOW	F476 F477	

PARAMETER	INPUTS	COMMENTS/LIMITS
RHR FLOW	F626	
RHR HEAT EXCHANGER INLET TEMPERATURE	T627	
RHR HEAT EXCHANGER OUTLET TEMPERATURE	T630	DCR TO MOVE LOWER LIMIT BELOW 100°F
CONTAINMENT SUMP LEVEL	N/A	
CONTAINMENT HYDROGEN CONCENTRATION	ALL OF THESE ARE NOT INPUT YET.	
CONTAINMENT HUMIDITY		
CONTAINMENT TEMPERATURE		
RWST LEVEL		
CST LEVEL		

DIAGRAM FOR DETERMINING RCS PRESSURE, TEMPERATURE LIMITS



RCS PRESSURE LIMITS - DETERMINED BY THE HEIGHT OF THE ACCEPTABLE REGION AT RCS AVERAGE TEMPERATURE

RCS COLD LEG TEMPERATURE LIMITS - DETERMINED BY THE WIDTH OF THE ACCEPTABLE REGION AT RCS PRESSURE

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