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ACCESSION NBR:8307270452 DOC.DATE: 83/07/01 NOTARIZED: NO DOCKET #
FACIL:50-397 WPPSS Nuclear Project, Unit 2, Washington Public Powe 05000397
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SUBJECT: Forwards rept on safety parameter display sys safety analysis,per 830415 commitment & response to Generic Ltr 82-33,Suppl 1 to NUREG-0737, "Requirements for Emergency Response Capability."

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July 1, 1983
G02-83-596

Docket No. 50-397

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Licensing Branch No. 2
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Schwencer:

Subject: NUCLEAR PROJECT NO. 2
SAFETY PARAMETER DISPLAY SYSTEM (SPDS)
SAFETY ANALYSIS REPORT, SUBMITTAL OF

Reference: G02-83-346, G. D. Bouchey (SS) to A. Schwencer (NRC), "Nuclear Project No. 2, Emergency Response Capability" dated April 15, 1983

The reference provided the Supply System response to NRC Generic Letter No. 82-33 (Supplement 1 to NUREG-0737 - Requirements for Emergency Response Capability) and committed to provide the subject report by July 1, 1983. The attached report fulfills that commitment.

Section 4.2 of Generic Letter 82-33 (Documentation and NRC Review) describes options for implementation of the SPDS with or without prior approval by the staff. As stated in the attachment to the referenced letter, due to the completion status and training commitments of the SPDS, the NRC pre-implementation review option proposed in 82-33 is not viable. Further, section 4.2 states that "if the changes are to be implemented without prior NRC approval, the Licensee's analysis shall be submitted to NRC promptly on completion of review by the Licensee's offsite safety review committee". The Supply System offsite review committee is the Corporate Nuclear Safety Review Board (CNSRB). The implementation of the SPDS on WNP-2 does not represent changes to an operating plant (with inherent safety implications) nor does review of the system fall within the scope of review responsibilities of the CNSRB as described in the standard technical specifications (section 6.5.2). The system design, review and construction has been conducted in the same

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manner as any design change: design reviews, design specifications, purchase specifications and construction following and QA programs. Safety related system review by the offsite review committee has not been imposed on the industry in the past. To expand the scope of the offsite review committee to cover the non-safety related SPDS system at WNP-2 is not justified. Therefore, due to the extensive review the SPDS has received to date and the scope of responsibility of the CNSRB as described in the standard technical specifications, review of the SPDS at WNP-2 by the CNSRB will not be performed.

Should you have any further questions, please contact Mr. R. M. Nelson, Manager, WNP-2 Licensing.

Very truly yours,

GD Bouchey

G. D. Bouchey
Manager, Nuclear Safety and Regulatory Programs

PLP/tmh
Enclosure

cc: R Auluck - NRC
WS Chin - BPA
A Toth - NRC

PDIS SAFETY ANALYSIS REPORT

INDEX

PDIS SAFETY ANALYSIS REPORT

1.0 INTRODUCTION

This report documents the basis for selection of the parameters utilized in the Safety Parameter Display System (SPDS) to assess the safety status of the reactor during abnormal or accident conditions. As requested by NRC's General Letter 82-33,[1] this document represents WNP-2's SPDS Safety Analysis Report.

At WNP-2 the system utilized to provide concise plant data is the Plant Data Information System (PDIS) and is used synonomously with SPDS in the following discussions. The primary impetus, following the TMI incident, was to provide operators an improved, concise data presentation for response to accident conditions. The additional requirements of a Technical Support Center (TSC), an Emergency Operations Facility (EOF), and the addition of a degreed engineer to the operating staff (STA) necessitated further the need for a concise, yet comprehensive, data information gathering and display tool. The primary purpose of the PDIS is to aid the operating crew in the control room in monitoring the status of the critical safety functions that constitute the basis of the WNP-2 Emergency Operating Procedures.

Due to the construction phase of WNP-2 while these requirements were being formulated, an integrated system (PDIS) has been developed and is currently being implemented that satisfies the SPDS (PDIS) requirements and is directly related to the Emergency Operating Procedures (EOPs), developed during this period. Therefore, the purpose of the PDIS is to provide comprehensive, concise, plant data to (in order of priority):

1. The Reactor Operators,
2. The Shift Techncial Advisor,
3. The Techncial Support Center Personnel, and,
4. The Emergency Offsite Facility Personnel.

A brief description of the PDIS hardware and interfaces is presented in Section 2.0. More detailed description is presented in the WNP-2 FSAR Sections 7.5.1.5, 7.5.1.6, 7.5.1.15, 7.7.1.23, and Appendix B, Section I.D. 2.

Also contained in Section 2.0 is a discussion of the PDIS software (see Subsection 2.3). The graphic portion of the PDIS displays critical safety parameter values in three levels of detail, utilizing both bar and trend graphs. The relationship between the Critical Safety Functions (CSFs) (see Subsection 3.2) and the displayed parameters is summarized in Table 1.0-1. The three displayed levels, i.e., Levels 1, 2, and 3, correspond to increasingly more detailed plant parameter status information. This tiered approach allows initial overall evaluation based on the critical safety function and follow-up actions based on specific parameters of interest. The levels displayed are as follows:

Level 1: Overview of parameters representing the 5 critical safety functions in bar chart format.

Level 2: Detailed specific safety parameters for each of the 5 CSFs in bar and trend chart format. Also, some parameters are displayed in mimic format.

Level 3: Contains Emergency Operating Procedures graphs and aids as contained in the EOPs.

Section 3.0 contains the basis for the parameter ranges and setpoint selection for the PDIS. This section constitutes the safety evaluation portion of this report.

This conceptual description of the graphic display system is made more implicit in the following sections.

The interrelationships of the PDIS with EOPs, in particular, as well as with operator training, RG 1.97, and the Detailed Control Room Design Review (DCRDR) is fully recognized and appreciated at WNP-2 and discussed in Section 4.0. Although not a requirement for the safety evaluation of PDIS parameters, this discussion completes the in-direct factors considered in determining the parameters selected for input into the PDIS.

TABLE 1.0-1

GRAPHIC DISPLAY SYSTEM PARAMETER OVERVIEW

Critical Safety Function	GDS Level 1 Display	GDS Level 2 Display
1. REACTIVITY CONTROL	o Neutron Flux - Bar Graph	o Neutron Flux - Bar Graph - Trend Graph o Scram Signal Sent - Status Light o Control Rods Full In - Status Light o Period - Algorithm With Readout
2. REACTOR CORE COOLING	o RPV Level, Wide Range - Bar Graph	o RPV Level - Bar Graph of Upset Narrow Wide, and Fuel Zone Ranges - Trend & Mimic of Wide Range o ECCS, Status Lights o MSRV & MSIV Mimic
3. COOLANT SYSTEM INTEGRITY	o RPV Pressure - Bar Graph	o RPV Pressure - Bar Graph - Trend Graph o Dry Well Pressure - Bar Graph - Trend Graph o Equip. Drain Sump Flow - Bar Graph - Trend Graph - Isol. Status o Floor Drain Sump Flow - Bar Graph - Trend Graph - Isol. Status o MSRV - Status Lights o MSIV - Status Lights o Total Sump Flow - 24 Hr. Readout

TABLE 1.0-1 (contd.)

Critical Safety Function	GDS Level 1 Display	GDS Level 2 Display
4. CONTAINMENT INTEGRITY	o Drywell Pressure - Bar Graph	o Drywell Pressure - Bar Graph - Trend Graph o Drywell Temperature - Bar Graph - Trend Graph o Suppression Pool Temp. - Bar Graph - Trend Graph o Suppression Pool Level - Bar Graph - Trend Graph o Containment Isol. Groups - Status Lights
5. RADIOACTIVITY CONTROL	o Elevated Release Activity - Bar Graph	o Elevated Release Activity - Bar Graph - Trend Graph o Post LOCA Containment Activity - Bar Graph - Trend Graph o Area Radiation Monitors - Alarm Status & Location Description

2.0 PDIS SYSTEM DESCRIPTION

2.1 Overview - Hardware

The PDIS hardware is part of the Prime C750 Computer Configuration as shown in Figure 2.1-1. Those components indicated by a cross-line border in the figure constitute the PDIS.

The normal signal path is through the Transient Data Acquisition System (TDAS), then through one of the two parallel interfaces to the Prime C750 and out to one of the PDIS subsystems. If the selected interface path between TDAS and the Prime C750 should fail or transmit erroneous data, the link shall automatically transfer to the other signal path. The PDIS uses computer driven colorgraphic Cathode Ray Tubes (CRT) to provide the users with plant data useful to their job scope. The PDIS is primarily for post accident use as an aid in making rapid assessment of the plant's safety status; however, it will also provide information to the plant operators during all modes of reactor operation. The users during post-accident situations are plant operators, shift technical advisors, technical support personnel, and corporate engineering personnel. The system is designed such that all users have their own information system capabilities at their own work stations.

The Transient Data Acquisition System (TDAS) and the Process Computer (PC) receive and process plant data, while the Prime C750 computer interacts with each of these components to provide the appropriate information to the PDIS. The TDAS, PC, Prime C750 software capabilities and system hardware requirements are defined in References 2 through 7.

The PDIS hardware consists of:

Control Room

- o Two Colorgraphic CRTs with Tough Pads
- o Two Matrix Printers
- o Line Printers
- o One Dedicated I/O Colorgraphic Terminal*
- o One B/W Terminal
- o Printer/Plotter
- o One STA Colorgraphic Terminal*

Technical Support Center

- o Two Colorgraphic Terminals*
- o One B/W Terminal
- o One Matrix Printer
- o Two Slave Monitors

Emergency Operations Facility

- o Two Colorgraphic Terminals*
- o One B/W Terminal
- o One Matrix Printer

* Note: A colorgraphic terminal consists of a CRT, a controller, and a keyboard

The four subsystems of the PDIS that utilize the above equipment are (a) the Graphics Display System (GDS) in the control room, (b) the Shift Technical Advisors Information System (STAIS), also in the control room, (c) the Techncial Support Center Information System (TSCIS) in the TSC, and (d) the Emergency Operations Facility Information System (EOFIS) located at the Techncial Data Center in the EOF. The comprehensive, concise, and useful information available on the GDS and common to all four subsystems constitute the key parameters necessary to represent the safety status of the plant. The software is reviewed generically in Section 2.3 of this report and the bases for parameter, range and setpoint selection for inclusion in the GDS is presented in Section 3.0.

Signals are hard-wired from either the Control Room or remote locations through cable trams to TDAS remote modules (multiplexers). (See Figure 2.1-1) The remote modules provide:

1. Electrical Isolation,
2. Signal conditioning,
3. Analog to digital conversion,
4. Multiplexing of the input data,
5. Interface to fiber-optic cables.

The signal output from the remote modules is transmitted through fiber-optic cables to the TDAS central control unit. The fiber-optic cables are inherently isolation devices. All components which interface with Class IE circuitry (i.e., isolation devices in the remote units) to extract signals are qualified according to the requirements of Regulatory Guide 1.89 and IEEE 323-1974 and IEEE-344-1975. The GDS subsystem of the PDIS, consisting of two 19 inch diagonal colorgraphic rack-mounted CRTs and two associated touch-pads, are mounted in the vertical portion of the P601 and P602 Panels in the main control room. The interface control for the CRTs is provided by a limited function touch-pad mounted beside them. The CRTs, touch pads, and controls are seismically mounted to prevent them from damaging surrounding instrumentation and controls during an earthquake initiated event.

2.2 Availability

The PDIS has an availability goal of greater than 99%. This is a goal, not a requirement, since the PDIS is an aid to critical system function monitoring and is not the sole source of information available to the reactor operators, i.e. the PDIS is not essential to the execution of symptom-oriented EOPs.

The usefulness of the PDIS is recognized, however, and the high availability goal is to be achieved by purchasing high quality, easily maintainable, reliable equipment. Sufficient spare parts, well trained staff of technicians, and accessible locations shall also be available to ensure rapid repair. Availability will be evaluated on an annual basis and changes to equipment and/or software will be made accordingly if the availability goal is not accomplished.

C750 COMPUTER CONFIGURATION

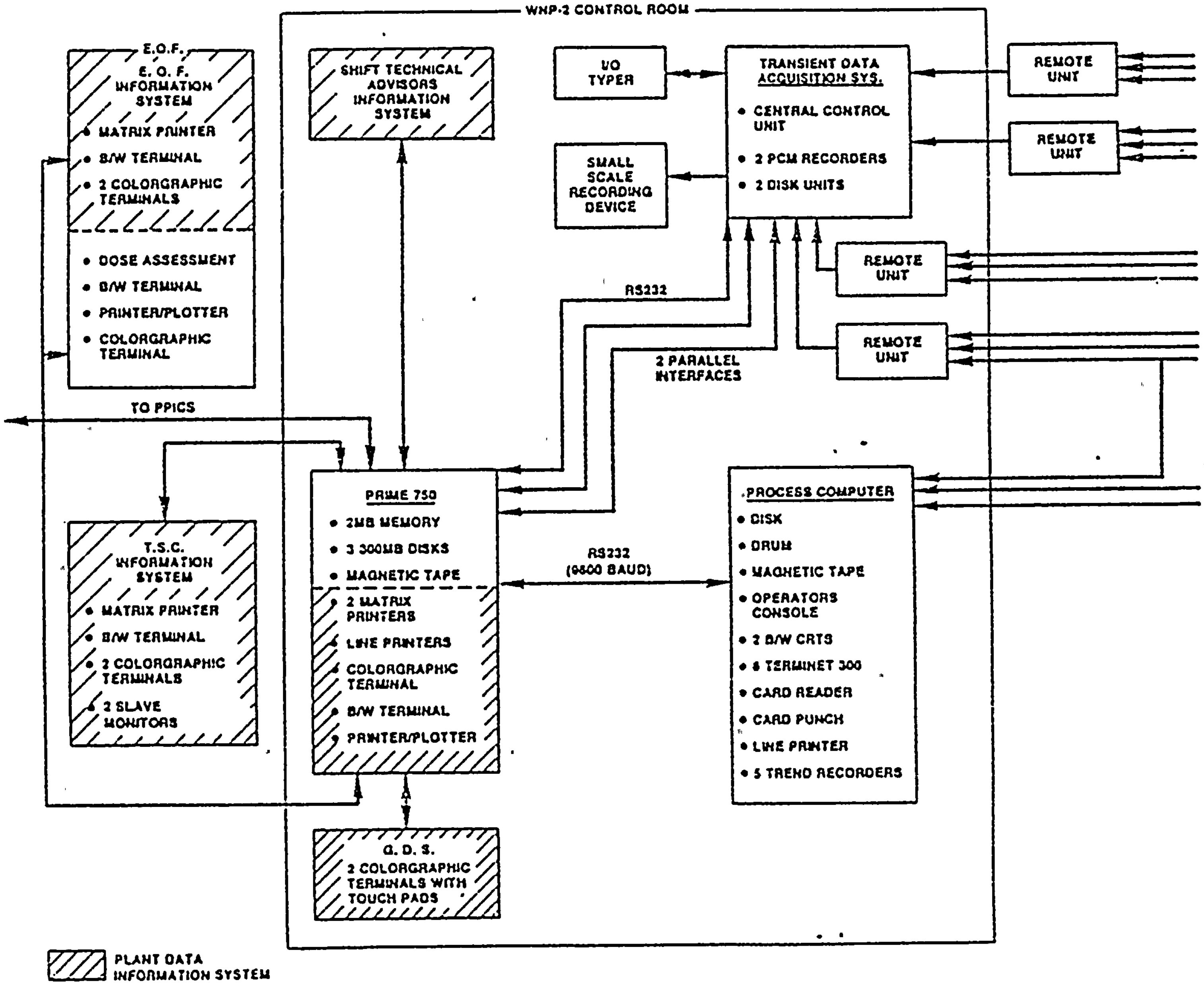

FIGURE 2.1-1

2.3 Overview - Software

System software security is protected against manipulation by unauthorized personnel through the use of passwords and keylocks. It is further protected against destruction by power losses. All corrections and improvements are reviewed and tested in accordance with plant procedures.

Data storage and recall capability is provided for the accident assessment data set from TDAS. A list of parameters and range of values is presented in Table 2.3-1 (FSAR Table 7.5-1 reproduced here). This is not a complete list of parameters available on TDAS, but represents those parameters included to satisfy Regulatory Guide 1.97 and post-accident monitoring. At least 2 hours of pre-event data and 12 hours of post-event data is recorded. Capacity to record at least two weeks of additional post-event data with reduced time resolution is provided. Archival data storage and the capability to transfer data between active memory and archival data storage without interrupting data acquisition and displays is provided.

The Graphic Display System (GDS) subsystem of the PDIS provides the comprehensive, concise information to be used as an aid in responding to emergency plant conditions. This subsystem is common to the control room, TSC and EOF. The GDS parameters are the functions that will occupy the body of this report and whose bases are discussed in the next section.

The basic set of displays used to meet the requirements of NUREG-0696 were developed by the BWR Owner's Groups[8] and validated by several operators at the "Perry" simulator in Tulsa, Oklahoma.[9] The display format is provided in three levels. Level 1 provides an overview status of the following critical safety parameters (CSF):

- o Reactivity Control
- o Reactor Core Cooling
- o Coolant System Integrity
- o Containment Integrity
- o Radioactivity Control

Level 2 provides for more detailed information of the above groups in two formats: (1) current status bar charts, and (2) a time trend of the various parameters. In addition, a time derivative (rate of change) of selected parameters is also provided.

Level 3 provides computer-generated data, mostly in graph format, to be used as an aid to the operator in implementing the emergency operating procedures. The displays are designed such that if a critical safety function (CSF) (see Section 3.2 for definition of CSF) exceeds its setpoint, a key parameter box will change colors to warn the operator. The operator would then press the function button associated with that parameter and receive a display of all the inputs to that CSF. Viewing this Level 2 display, the operator can quickly identify which of these inputs is out of limits by observing the yellow or red color in the bar chart display of each input parameter. Since the five CSF parameter boxes are incorporated at the bottom of all displays, the operator can tell the status of all CSFs no matter which display he has on the CRT.

When equivalent signals are present, the GDS signals are validated prior to processing on a real time basis by comparing the displayed signal with a companion signal. If a companion signal is not available, one is generated using a combination of other signals, if possible. If the signal validation is not achieved, the operator is informed by the words "INVALID SIGNAL" displayed either in the bar for bar graph displays or in the trend graph for trend graph displays. For those signals that do not provide an input to a bar or trend graph, but are used for display derivation (e.g. LPCS flow in Core Cooling bar chart), the invalid signal point ID shall be displayed at the bottom of the screen in red text.

TABLE 2.3-1

SAFETY-RELATED DISPLAY INSTRUMENTATION

Design Critieria	Type Readout	Number of Channels	Range	Type & Category[1]	Display Instrument Accuracy	Location
1. Reactor Vessel Pressure	Recorder	2	0-1500 psig	A,1	± 2% FS	CR
2. Reactor Vessel Water Level	Recorder	2	-150"/0/+60"	A,1	± 2% FS	CR
	Recorder	2	-117" - -317"	A,1	± 2% FS	CR
3. Neutron Flux Power Level (SRM)	Recorder	4	10^{-7} - 10^{-3}% Power[2] 10^{-3} - 10% Power[3]	A,	± 2%	CR
4. Main Steam Line Flow	Indicator	4	0 - 4.25 X 10^6 lb/hr	---	± 2% FS	CR
5. RCIC Flow	Indicator	1	0 - 700 GPM	D,2	± 2% FS	CR
6. RCIC Discharge Pressure	Indicator	1	0 - 1500 psig .	D,2	± 2% FS	CR
7. HPCS Flow	Indicator	1	0 - 8000 GPM	D,2	± 2% FS	CR
8. HPCS Discharge Pressure	Indicator	1	0 - 1500 psig	---	± 2% FS	CR
9. LPCS Flow	Indicator	1	0 - 10,000 GPM	D,2	± 2% FS	CR
10. Drywell Atmos & Suppress Atmos Temps	Indicator Print Out	2	50 - 400°F	D,2	± 2% FS	CR
11. LOCA Radiation High Range Area Monitors	Recorder	2 2	1 - 10^8 R/hr	C,E,2	± 2% FS	CR
12. Leak Detection Radiation Monitors	Recorder	2 2	10^0 - 10^6 CPM 10^0 - 10^6 CPM	--- ---	± 2% FS ± 2% FS	CR CR

NOTE: 1: The instruments meet the recommendations required by the Category type as described in Regulatory Guide 1.97, Revision 2.

2: Inserted

3: Withdrawn

TABLE 2.3-1 (Cont'd)

Design Critieria	Type Readout	Number of Channels	Range	Type & Category[1]	Display Instrument Accuracy	Location
13. Primary Containment Hydrogen Concentration	Recorder	2	0 - 30%	C,1	± 2% FS	CR
14. Primary Containment Oxygen Concentration	Recorder	2	0 - 10%	C,1	± 2% FS	CR
15. Suppression Chamber Pressure	Recorder	2	0 - 60 psig	D,2	± 2% FS	CR
16. Suppression Pool Temperature	Print Out Indicator	2	30 - 230°F	D,2	± 2% FS	CR
17. Suppression Pool Water Level	Recorder	2	+25"/0/-25"	C,1	± 2% FS	CR
18. Bldg. Gaseous Release Monitor	Recorder	3	10^1 - 10^6 CPM	E,C,2	± 2% Span	CR
		3	10^1 - 10^6 CPM	E,C,2	± 2% Span	CR
		2	10^{-2} - 10^4 R/hr	C,2	± 2% Span	CR
19. Containment Instrument Air	Indicator	2	0-150 psig	D,2	± 2% FS	CR
20. Wind Speed	Recorder	1	0 - 67 MPH	E,3	± 2% FS	CR
Wind Direction	Recorder	1	0° - 360°	E,3	± 2%	CR
21. Temperature Differential	Recorder	1	±15°F	E,3	± 2%	CR
22. Radiation Exposure Rate	Recorder	3	10^{-2} - 10^4 R/hr	C,2	± 2% Span	CR
23. SRV Position Indication	Indicator	18	Full Closed to Full Open	D,2	-----	CR
24. Power Supply Monitoring	Voltmeter	6	0-5.25 kVAC	D,2	± 2% FS	CR
		4	0-600 VAC	D,2	± 2% FS	CR
		3	0-300 VAC	D,2	± 2% FS	CR
		5	0-150 VAC	D,2	± 2% FS	CR
		4	±30 VAC	D,2	± 2% FS	CR
		29	DC Ammeters of various ranges			CR

TABLE 2.3-1 (Cont'd)

Design Critieria	Type Readout	Number of Channels	Range	Type & Category[1]	Display Instrument Accuracy	Location
25. Feed Water Flow	Indicator	2	0 - 8.5 X 10^6 #/hr	D,3	± 2% FS	CR
26. CST Level Indicator	Indicator	2	0 - 35 ft.	D,3	± 2% FS	CR
27. RHR Flow (LPCI and Shut-down Cooling)	Indicator	2	0 - 10,000 GPM	D,2	± 2% FS	CR
(Head Spray)		1	0 - 600 GPM	D,2	± 2% FS	CR
28. RHR HX Outlet Temperature	Recorder	2	0 - 600°F	D,3	± 2% FS	CR
RHR Service Water Flow	Indicator	2	0 - 10,000 GPM	D,2	± 2% FS	CR
29. SLCS Flow Rate	Indicator	1	0 - 50 GPM	D,2	± 2% FS	CR
30. SLCS Tank Level	Indicator	1	0 - 5000 Gal	D,2	± 2% FS	CR
31. SSW System Pump Discharge Line Pressure	Indicator	2	0 - 300 psig	---	± 2% FS	CR
		1	0 - 100 psig	---	± 2% FS	CR
32. SSW System Flow Rate	Indicator	2	0 - 12,000 GPM	D,2	± 2% FS	CR
HPCS SS Flow Rate	Indicator	1	0-1320 GPM	D,2	± 2% FS	CR
33. SSW Pond. Water Level	Indicator	4	0 - 20 ft.	---	± 2% FS	CR
34. Spent Fuel Pool Cooling	Indicator	3	0 - 212°F	---	± 2% FS	CR
35. Main Control Room Temper-ature*	Indicator	2	50 - 100°F	---	± 2%	CR
36. SGTS Flow Rate	Indicator	4	0 - 6000 CFM	---	± 2%	CR
37. CAC System Flow Rate	Indicator	4	0 - 300 CFM	---	± 2%	CR

*Not included in TDAS.

TABLE 2.3-1 (Cont'd)

Design Critieria	Type Readout	Number of Channels	Range	Type & Category[1]	Display Instrument Accuracy	Location
38. MSIV-LCS Outboard Steam Line Header Pressure	Indicator	1 1	0 - 2 psig 0 - 50 psig	D,2	± 2% FS ± 2% FS	CR CR
39. MSIV-LCS Steam Line Pressure Between MSIVs	Indicator	1 1	0 - 8 psig 0 - 50 psig	D,2	± 2% FS ± 2% FS	CR CR
40. MSIV-LCS Leakage Flow	Indicator	4	0 - 1 CFM	---	± 2% FS	CR
41. Radioactive Tank Levels	Recorder Indicator	6 5	0 - 100% 0 - 100%	D,3 D,3	± 2% FS ± 2% FS	RW Bldg. RW Bldg.
42. Primary Containment Moisture*	Recorder	2	0 - 100% RH	---	± 2% FS	CR
43. Primary Containment Pressure	Recorder Recorder	2 2	Pen #1 0 - 25 psig Pen #2 0 - 180 psig -5 to +3 psig	A,B,1 A,B,1 A,B,1	± 2% FS ± 2% FS ± 2% FS	CR CR CR
44. Emergency Ventilation Damper Position	Indicator	1 ea.	Open-Close	D,2	---	CR
45. Control Rod Position	Indicator	1 ea.	Full in or not full in	B,3	---	CR
46. Primary Containment Isolation Valve Position	Indicator	1 ea.	Closed or not closed	B,1	---	CR
47. Graphic Display System	CRT	2	N/A	---	N/A	CR

*Not included in TDAS.

3.0 PDIS PARAMETER SELECTION

3.1 Reactor Modes

The PDIS is available during normal power operation, hot standby, and (non-refueling) cold shutdown periods to confirm the CSFs are being maintained and to indicate when their status is in question. By defining these reactor operating modes as initial conditions, the parameter value ranges are determined for display input.

3.2 Critical Safety Functions (CSFs)

CSFs are defined as those limited number or minimum set of parameters or derived variables from which the plant safety status can be inferred. For the WNP-2 GDS Level 1 display, these CSFs are:

- o Reactivity Control
- o Core Cooling
- o Coolant System Integrity
- o Containment Integrity
- o Radioactivity Control

The basis for this CSF parameter set is the fact that these functions are the same or equivalent to the CSFs used in the symptom-based, WNP-2 Emergency Operating Procedures (EOPs). Therefore, the PDIS/GDS users have a direct correlation for performing their roles in the execution of the EOPs. The underlying basis for these CSFs is, of course, the principle of 'defense in depth'. For protection of the fuel clad barrier, one must status the reactivity (power) and the core cooling. Coolant system integrity provides information on the second barrier, the reactor coolant boundary. Similarly, the containment integrity function provides information on the third barrier, primary containment boundary and penetrations. Lastly, radioactivity control function

provides information necessary for the last protective barrier, movement or sheltering of people if a major release should occur. Radioactivity control function also provides information on leakage paths prior to release.

Based on these 5 or equivalent CSFs, the EOPs were developed by the BWR Owners Group in which WNP-2 personnel participated. These 5 CSFs also then became Level 1 for the GDS display. Bases for individual parameters (Level 2 displays) within the CSF groups are given in the following section (3.3).

3.3 Design Basis

Using the CSFs defined in Section 3.2 to develop the EOPs then specified those parameters as the basis for the Graphics Display System. The Generic Emergency Procedure Guidelines, as developed by the BWR Owners Group, were evaluated by a BWR Owners Subcommittee and Sandia Laboratories. From this review, a basic set of displays were developed for the GDS. This extensive evaluation, which started in early 1981, is documented in two reports developed in conjunction with the Department of Energy and Sandia Laboratories (References 8 and 9).

The WNP-2 Graphics Display System (GDS) displays and algorithms are based on the BWR Owners Group Development Program described above. The displays and the site specific design basis for each CSF are given in more detail below, particularly for Level 2 setpoints and ranges. The philosophy of color change is based on the severity of plant parameters. If a condition exists where either manual or automatic action is required to prevent plant safety from degrading, then the abnormal parameter shall change to yellow. If a safety parameter is about to degrade to a more severe safety state, then it shall change to red. This philosophy is common to all GDS displays.

3.3.1 Reactivity

Reactivity is not a directly measured parameter, so the selected parameters for Level 2 indication of reactivity control are:

	Range
APRM Reactor Power	0-125%
Reactor Period	$0-\pm\infty$
Control Rods Full In	NA

Reactor power and period provide a direct indication of reactivity control. The primary reactivity control system in a BWR is the Control Rod Drive (CRD) system. Therefore, the "Control Rods Full In Following Scram" signal is annunciated to verify the CRD has performed its function.

Bases for Range and Setpoint Values

APRM Reactor Power - The WNP-2 Safety Analysis shows that any reactor power excursion above 125% rated power is immediately terminated by Doppler and negative void reactivity feedbacks. Even those transients where the CRDs fail to insert, i.e., ATWS events, result in reactor power levels less than 50% of rated power. Therefore, a range of 0-125% of rated power is sufficient for accident monitoring. The display setpoints were determined as follows. The first indicator turns yellow when reactor power is greater than or equal to 118% of rated power. This is the technical specification value for a reactor scram on high flux. The second indicator will turn red if a scram signal is sensed and reactor power is still above 3% of rated power after 10 seconds from scram signal. The red indicator is therefore indicative that the CRDs did not fully insert. The 10 second delay allows for rod insertion time and some delayed neutron decay.

Reactor Period - The reactor period measurement is useful in the startup modes to indicate abnormal reactivity additions prior to where the reactor power measurements become sufficiently sensitive. The count rate used in the algorithm comes from the highest rate of the two SRMs which have a range of 10^{-1} to 10^{6} cps. The algorithm used is:

$$\text{Period} = \frac{t}{\text{Ln}\ {}^{C}/_{Co}},$$

Where t = time between measurement of Co and C is equal to 1 second.

C = largest value of two SRM channels after 1 second

Co = largest value of two SRM channels at time zero.

The setpoint for the yellow indicator is 5 seconds which is still a stable period and far from a prompt critical situation but is indicative of a rapid reactivity insertion condition. There is no second or red indicator necessary for reactor period since feedback effects would terminate the transient prior to any reasonable operator action.

Control Rods Full In - This indicator uses an on/off logic so range is not applicable. The logic states that if the rods full in signal is not received after 7 seconds of receiving 2 out of 4 RPS channel trip signals, then the indicator turns red. No first (yellow) indicator is used. Seven seconds is chosen to allow sufficient rod travel time under potentially adverse conditions.

3.3.2 Core Cooling

In BWRs the core cooling is directly related to reactor water level. As long as the core remains greater than two-thirds covered, the core is adequately cooled. Therefore, the GDS utilizes several water level measurements and indications for the Level 2 display of core cooling. In addition, status boxes of the various ECCS systems are provided.

Bases for Range and Setpoint Values

WNP-2 has multiple water level measurement instruments covering ranges from bottom of active fuel (BAF) to overfilling the vessel. The name and ranges of the instruments utilized for the GDS are:

Upset Range	0 - 180 inches
Narrow Range	0 - 60 inches
Wide Range	-150 - 60 inches (1 of 2 channels)
Fuel Zone Range	-317 - (-)117 inches (1 of 2 channels)

Where instrument zero corresponds to about the bottom of the steam separator skirts, -317 inches is below BAF, and 180 inches is above the steam piping outlet. Therefore, the range for Level 2 GDS display is adequte. The setpoints for the color indicators have been set such that when the water level drops to -50 inches (L2), the yellow indicator is lit. L2 corresponds to the ECCS actuation setpoint and is consistent with the philosophy of using yellow or first indicator for a setpoint denoting action is required. If the water level continues to drop to -129 inches (L1), then the indicator turns red indicating further degradation and that additional safety action is necessary, e.g., ADS may be required.

The core cooling indicator will also change to yellow if the reactor water level is about to exceed the upper limit of the normal water level instrument, i.e., +54.5 inches (L8). Although not an indication of core cooling problems at the moment this happens, the operator needs to be aware of excessively high water level to prevent filling of the main steam lines with water.

In order to assist the reactor operator in determining if the proper safety function was initiated, the Level 2 GDS display contains a status light indicator for each of the ECCS systems. These systems are the HPCS, ADS, LPCS, LPCI-A, B and C. The status indicator for each system is normally black, changing to red when activated and flow is present.

3.3.3 Coolant System Integrity

The integrity of the reactor coolant system pressure may be jeopardized in several different ways. If the reactor vessel pressure exceeds the design pressure, leaks or pipe ruptures may occur. Also, if SRVs or MSIV are open when not required, this constitutes a break in the RCS boundary; and finally, small leaks from the RCS constitute potential loss of RCS pressure boundary. The GDS utilizes both direct and in-direct measurements to assess the integrity of the reactor coolant system boundary. The reactor coolant pressure is monitored to indicate overpressurization, the drywell floor and equipment drain flows are measured and displayed to indicate leakage, the primary containment pressure is displayed since this can indicate leakage of reactor coolant, (high drywell pressure may also be indicative of loss of containment cooling), if the reactor pressure is low but an SRV is open this is indicated, and if a pair of MSIVs in the same steam line are open when required to be closed, this is also indicated.

Bases for Range and Setpoint Values

RPV Pressure - The reactor vessel pressure range displayed is 0 - 1500 psig. WNP-2 FSAR Chapter 15 and 5 transients show the peak reactor vessel pressure as less than 1250 psig for all cases. The peak system pressures indicated for ATWS events is less than 1300 psig. Therefore, a range of 0 - 1500 psig is sufficient. The first (yellow) indication setpoint is 1148 psig which represents the lowest SRV setpoint in the safety-relief mode. This setpoint was chosen since it represents the point at which the relief mode of the SRVs cannot stop the pressure increase and the safety mode is being relied upon. The second (red) indication setpoint is set at 1250 psig, which represents the design pressure. Exceeding this value indicates severe stress is placed on the vessel and additional safety actions are necessary.

Drywell Pressure - This is an indirect indicator and is included in the Level 2 display for information only. The range is 0 - 25 psig to be sensitive at the lower end of range, i.e., the high containment pressures are not of much interest in checking if reactor coolant boundary is intact. No setpoints are associated with this indicator as no operator action will be keyed to this parameter for the coolant system integrity CSF.

Drain Sump Flows - Level 2 displays both the equipment drain (identified leakage) sump flow with a range of 0 - 30 gpm and the floor drain (unidentified leakage) sump flow with a range of 0 - 6 gpm. In addition, if the isolation valves associated with the equipment or floor drain sumps are closed that will be indicated by "system isolated" display text. The range of values are sufficient in that technical specification actions must be taken before exceeding the values of the ranges. No setpoint is associated with the equipment drain sump flow since this represents known or identified leakage, e.g., from pump seals. It is included as an indicator only and to help estimate its contribution to the integrated sump flow for the last 24 hours which is also displayed. If the unidentified flow, i.e., the floor drain sump flow exceeds 5 gpm, the indicator turns yellow. This setpoint corresponds to the Technical Specification requiring reduction of the leakage or bringing the plant to shutdown. Similarly, the integrated sump flow indicates yellow if the 24 hour accumulated flow is 36,000 gallons (25 gpm averaged over 24 hours) per the same technical specification.

MSIV Closure - Parameter range is not applicable since this display signal is a logic display. The MSIV position logic* is based on the following:

1. If all MSIVs open display "OPEN" and RED.

2. If all MSIVs closed display "SHUT" and GREEN.

SRV Closure - Parameter range is not applicable since this display signal is a logic display. The SRV position logic is based on the following:

1. If the SRV position is shut, display "SHUT" and green.

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*A MSIV close signal is generated by reactor water level at L2 (-38 inches), drywell pressure high (1.69 psig), main steam line flow high (104 psid) or low pressure (825 psig) or high radiation (2.5 times background), main steam line tunnel temperature high (145^{o}F.) or delta T high (50^{o}F.), condenser vacuum low (23" Hg), and high radiation.

2. If the reactor vessel pressure is less than 900 psig and any SRV is open, display "OPEN" and red.

3. If the reactor vessel pressure is greater than 900 psig and any SRV is open, display "OPEN" and red.

The SRV positions are monitored by acoustic monitors to indicate open or closed. The reactor vessel pressure is monitored as described in Section 3.3.3 above. The 900 psig logic setpoint for the GDS display to change from green to yellow is based on the operating procedures requirement to maintain pressure control limits of between 930 psig and 1075 psig in the event of a pressure transient. The difference between 900 psig and 930 psig is for operator margin, considering instrument readings, reaction time and system time constants.

3.3.4 Containment Integrity

The primary containment system, which includes the pressure suppression pool, provides a barrier to prevent release of radioactive material to the environment and provides a heat sink to absorb energy released from the reactor. Threat to the containment integrity can result from overpressurization, under or negative pressure, and loss of containment isolation status. The parameters monitored and displayed to reflect status of the containment integrity are drywell pressure, drywell temperature, suppression pool temperature, suppression pool level, and containment isolation groups.

Bases for Range and Setpoint Values

Drywell Pressure - The drywell pressure measurement range is 0 - 180 psig. Since design pressure is 45 psig, this represents 4 times design pressure for range which is adequate. A supplemental pressure measurement with a range of -5 to +3 psig is also displayed. The original design pressure for a negative differential pressure was -4 psig which is covered by the supplemental pressure measurement. There are no mechanistic transients that could cause a negative differential pressure in excess of the -4 psig, hence the range is adequate. The Level 2 display setpoints are established such that the first (yellow) setpoint is at +1.69 psig or -2 psig. The 1.69 psig is based on analyses that show containment pressure will subsequently be limited to less than a total pressure of 34.7 psig following a DBA LOCA. In addition, a reactor trip will occur when the primary containment pressure exceeds 1.69 psig. The -2 psig will ensure adequate warning to the operator of vacuum problems in the containment. It also represents the minimum pressure attained on inadvertent operation of the containment sprays. The second (red) setpoint is set at the design pressure of 45 psig, providing the operator warning he has reached the design limit. Any further increase in pressure is beyond the design bases.

- Drywell Temperature - The drywell temperature measurement range is 50 to 400°F. This range adequately covers the range from normal operation (90°F) to greater than analyzed DBA LOCA temperature (340°F). A maximum limit of 135°F average bulk temperature has been set for normal operation to preclude exceeding 340°F during a DBA. Correspondingly, the GDS first (yellow) setpoint is set at 135°F, changing to the second (red) setpoint at 340°F.

Suppression Pool Temperature - A temperature measurement range of 30 to 230°F has been established for the suppression pool bulk temperature. A peak suppression pool temperature for DBA LOCA has been analyzed to be less than

220^{o}F with initial pool temperatures of 90^{o}F. Therefore, the parameter range adequately covers all analyzed scenarios. To allow for RCIC test conditions where suppression pool temperature will increase due to the test, the setpoint for the first (yellow) setpoint is established at 105^{o}F and the second (red) setpoint at 110^{o}F. Therefore, the 105^{o}F allows a minimum of 15^{o}F (i.e., 105 - 90 = 15^{o}F) in which normal testing can be performed without an unusual event or alert being declared. However, the 110^{o}F limit is established to (a) indicate that an abnormal condition exists, e.g., a stuck open relief valve, (b) allow sufficient margin such that a complete reactor blowdown from full power operating conditions can be performed without exceeding a suppression pool temperature limit of 200^{o}F and (c) if the reactor cannot be shutdown and the suppression pool temperature reaches 110^{o}F, then SLCS must be initiated.

Suppression Pool Level - The suppression pool water level measurement range is -25 to +25 inches, corresponding to normal reactor instrumentation.

To assure an adequate volume of water is present at the initiation of a LOCA, a lower first (yellow) setpoint has been established at -2 inches. Conversely, to assure an adequate air space in the wetwell so as not to contribute further to overpressurization and to limit structural loads on the containment from the dynamic forces during the initial clearing of the drywell downcomers in a DBA LOCA, an upper first (yellow) setpoint has been established at +2 inches. A 2 inch variation represents approximately 750 ft^{3} of water variation in the suppression pool which is achievable and would have minimal impact on the analysis.

Containment Isolation - Although not a threat to containment structural integrity, per se, containment isolation is a necessary function to help mitigate radiological releases and is valuable safety status information. There is no range of values as the status logic is a simple open (non-isolated)/ close (isolated) annunciation. The ten containment isolation valve groups are shown in Table 3.3.4-1. If a containment isolation valve is called upon to close, and does not, the group isolation status light changes from green to red.

TABLE 3.3.4-1

GDS CONTAINMENT ISOLATION GROUPS

Group	Description
1	Main Steam Line Isolation
2	Reactor Coolant Sample Line
3	Containment Vent and Purge Lines
4	Recirculation Hydraulic Lines Floor and Equipment Sump Drain Suppression Pool Cleanup Reactor Building Circulating Cooling Water Radiation Monitor Supply/Return Line Isolation
5	Residual Heat Removal Check Valve Return Line Bypass Valve (Normally locked closed)
6	Residual Heat Removal Isolation
7	Reactor Water Cleanup Isolation
8	RCIC Steam Supply
9	RCIC Turbine Exhaust Vacuum Breaker Isolation
10	RHR, LPCS, and HPCS Test Line Isolation Suppression Pool Spray Lines

3.3.5 Radioactivity Control

Radioactivity control is the ability to monitor activity that is being released or has a potential for release. This aids the operator and support personnel in defining offsite consequences, as well as providing a possible diagnostic tool in determining an accident's cause and/or progression. The primary path of release during an accident situation would be gases exiting the plant through the elevated release duct. All other release paths are isolated if high radiation exists. Since release of these gases indicate failed fuel and a leak from the primary containment or an unisolatable break in the reactor coolant pressure boundary, an alarm in this status box should alert the operator to evaluate the safety status of both the reactor coolant boundary and the primary containment. As an aid to the diagnosis, the Level 2 display contains the elevated release activity, post-LOCA containment activity, a listing of any area radiation alarms which might have gone off, and the Standby Gas Treatment System Flow Rate.

Bases for Range and Setpoint Values

The range of measurement associated with each of the displayed radioactivity monitors is as follows:

	Range
Post-LOCA Containment Activity	10^{0}-10^{8} R/hr
Elevated Release Activity	10^{-2}-10^{4} R/hr

Since the area radiation monitors will display in a list format if their setpoints are reached, i.e., an on/off logic, discussion of their range is not applicable. The parameter measurement ranges listed above are based on Reg. Guide 1.97 recommendations and indeed, meet or exceed those ranges.

Although the elevated release activity is the primary safety parameter for this CSF, the regulatory limits are based on, and correctly so, site boundary dose. Since the boundary dose defends not only on elevated release activity, but also on varying meteorological conditions, no fixed limit can be set for elevated release. The same is true for the containment activity monitors. The site boundary dose rate can be obtained from the Emergency Dose Projection System (EDPS) located at the same PDIS terminals as the GDS displays. Therefore, the radioactivity monitors are displayed without setpoints.

4.0 INTEGRATION

Although the primary purpose of this report is to provide the bases upon which the PDIS parameter ranges and setpoints were chosen, the integration of the PDIS with the development of symptom oriented Emergency Operating Procedures, with the operating staff training, with Reg. Guide 1.97 guidance, and with control room design review verifies that the interrelationships have been considered in the PDIS design. These interrelationships provide additional safety assurance in the evaluation of the assessment of the overall enhancement of the operator's ability to comprehend plant conditions and to cope with emergencies. Since the emergency response facilities (TSC and EOF) contain the same GDS display as the control room, any future upgrade to the control room GDS is automatically included in the TSC and EOF. This interrelationship of PDIS/GDS to EOF, TSC is therefore not discussed any further. It should also be noted, that the design of PDIS is not fixed. As the use of PDIS increases and its incorporation into the WNP-2 simulator takes place, the displays will be redefined to better serve its intended function.

4.1 Relation to EOPs

The PDIS and the Emergency Operating Porcedures (EOPs) are related in the most fundamental manner through the Critical Safety functions (CSFs). That is, the 5 CSFs are the bases from which the PDIS/GDS displays were chosen and the same 5 CSFs are the major or first line entry points into the EOPs. To illustrate this point further, consider the following table:

CSF	PDIS/GDS Level/Display	EOP Entry Procedure
Reactivity Control	Power Level (APRM)	5.1.3
Core Cooling	Water Level	5.1.1
Coolant System Integrity	RP Vessel Pressure	5.1.2
Containment Integrity	Drywell Pressure	5.2.3*
Radioactivity Control	Elevated Release Activity	Later

As illustrated by the footnote to the above table, Level 2 of the PDIS/GDS display presents additional parameters for each CSF providing a more complete picture of the reactor and containment status. As the operator completes each step of the EOP, he is then able to determine the effect or effectiveness of that action.

Level 3 of the PDIS/GDS display contains many of the limit curves referenced in the EOPs. This gives the control room, TSC and EOF personnel quick access to the operating limits applicable to a given situation.

4. Relation to Training

Being in the advantageous position of completing the Emergency Operating Procedures (EOPs) in a coordinated fashion with the PDIS/GDS completion, the WNP-2 operator training can also be done in a coordinated manner. The principle technique to be utilized is procedure/hardware walk-throughs. The WNP-2 control room walk-throughs by each rotating shift crew will demonstrate actions

*EOPs 5.2.1, 5.2.2 and 5.2.4 are for suppression pool temperature, drywell temperature, and suppression pool level, respectively. The drywell pressure was selected for GDS Level 1 display, but the procedure calls for 5.2.1 through 5.2.4 to be executed simultaneously.

required by EOPs can be related to equipment, control, and indications both using the Graphic Display Sytem (GDS) and without the GDS. The operator training is actually being accomplished in a three step process. First is familarization of the operators with the GDS display levels and touch pad operation. Secondly, the GDS, as installed in the control room, will be introduced as a functional unit during simulated plant operations. The third step is coordinated walk-through of EOPs, utilizing the GDS, and showing alternate signals if GDS is not available.

4.3 Relation to Reg. Guide 1.97

WNP-2 compliance to Reg. Guide 1.97 is documented in the WNP-2 FSAR, Section 7.5.2.3.e. The only unresolved issue is the inadequate core cooling issue which is being resolved by participation in the BWR Owners Group efforts to resolve this issue with the NRC. Compliance with Reg. Guide 1.97 ensures that appropriate parameters and parameter ranges are available for PDIS/GDS display. This was implicitly, and in some cases, explicitly, stated in the bases justifications in Section 3.3. In summary, development of the PDIS/GDS was integrated from the start with the WNP-2 effort for compliance to Reg. Guide 1.97.

4.4 Relation to DCRDR

The Supply System recently submitted our preliminary Control Room Design Review Report which will satisfy the program plan requirements of NUREG 0737, Supplement 1 (Reference 10). The PDIS as described in Section 2 of this report is an integral part of the control room instrumentation configured for easy appraisal and analysis combined with the GDS providing CRT displays of noted parameters. The PDIS/GDS is derived from the BWR Owners Group Control Room Subcommittee work described previously. Since the PDIS System has been completed prior to or in conjunction with the DCRDR, adequate assurances exist that control room design modifications affecting PDIS are incorporated in the PDIS design and conversely, any modifications to the PDIS/GDS will automatically be included as part of the DCRDR.

4.5 Coordination of Functions

At the conception of the WNP-2 TMI program, the responsibility for resolution and completion of each TMI concern was assigned to a lead Technical Reviewer, usually the system expert, or an individual having expertise in the specific area of concern, with overall program responsibility lying with the WNP-2 TMI Program Manager. The TMI Program Manager has also been assigned the responsibilities associated with the Emergency Response Facility Systems, particularly the GDS, and was assigned responsibility to coordinate these efforts and ensure integration of the TMI activities. Therefore, the Supply System has a "point" contact for the diverse activities associated with the EOF, TSC, DCRDR, Reg. Guide 1.97 and NUREG 0737 leading to the advance state of completion for these activities. It is worthy to note that the TMI Program Manager is also a Shift Technical Advisor, i.e., a principal user of the PDIS/GDS system.

5.0 SUMMARY AND CONCLUSIONS

The WNP-2 Plant Data Information System (PDIS)/Graphic Display System (GDS) performs the functions required of a Safety Parameter Display System (SPDS). The PDIS/GDS is an aid for use by the operating crew in monitoring the status of the Critical Safety Functions (CSFs). Instrumentation is provided in conformance with Reg. Guide 1.97 for monitoring CSF status which provides input to the PDIS/GDS. However, since PDIS/GDS is not a fully qualified, redundant system, it remains an aid to the operators, not their sole source for CSF monitoring.

The "lessons learned" from TMI indicated the importance of symptom-oriented Emergency Operating Procedures (EOPs) and the need for concise, yet comprehensive, monitoring of CSFs to aid in implementing the EOPs. The WNP-2 PDIS is based on the same, or equivalent, CSF utilized in deriving the EOPs, ensuring correlation between the two items. In addition, a review of the EOPs by the personnel involved in the PDIS/GDS design specifications further ensures proper interface of these two related functions. Other interface functions, e.g., with DCRDR and training, have been promoted by utilizing the WNP-2 TMI Program Manager as a focal point for these activities.

The bases for the ranges and setpoints for the CSF displays have been presented in this report and reflect the adequacy of their values. The bases also reflect that the interface with Reg. Guide 1.97 was carefully observed during the PDIS design.

It can be concluded that with proper operator training with symptom-based procedures and the PDIS function, WNP-2 has developed a comprehensive, concise, plant data source envisioned for proper SPDS implementation.

6.0 REFERENCES

1. NRC Generic "Supplement 1 to NUREG-0737 - Requirements for Emergenpability", dated December 17, 1982

2. Startup Traquisition Systems, Contract 2808-98, Technical Pı

3. Station NucManual, Volume 2; Process Computer; NEDE-24810,

4. Process ComDescription; WPPSS File #2-2C91-00-2-1

5. Primos Comm. Guide FDR3108-101 B File No. T2012 (Contract 9:

6. Shape MonitContract 2808-97

7. Transient Dn System Signal List, Burns and Roe Drawing E554

8. VuePoint Prual and Descriptive Literature File No. 1001A (Cont

9. Emergency Oaredness System Requirements, and Conceptual Design; K. McElroy, V. M. Lee, August 6, 1981

10. Letter G02-ouchey (Supply System) to A. Schwencer (NRC), "Nuc Control Room Design Review, Submittal of Preliminaryd April 14, 1983