FRAMATOME ANP

February 5, 2002
NRC:02:011

Document Control Desk
ATTN: Chief, Planning, Program and Management Support Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Responses to an Informal Request for Supporting Information

The NRC informally requested that members of the B&WOG provide supporting information concerning the matter of operator action to shut down the reactor coolant pumps following a core flooding line break. This request was made to ensure that appropriate emergency procedures are established to safely conduct this action within an appropriate time period.

To help expedite the NRC's review process, Framatome ANP requested that each member of the B&WOG provide answers to the set of NRC informal questions. These responses are enclosed.

If this information needs to be formally submitted on a docket, Framatome suggests that the individual utilities be contacted since this subject is plant specific and not generic to the B&WOG. In addition, the two utilities that are directly affected by these questions will appreciate a letter from the NRC indicating that its evaluation of the matter was satisfactorily concluded.

Very truly yours,

Gerald S Holm

James F. Mallay, Director
Regulatory Affairs

/lmk

Attachment

cc: J. S. Cushing
Project 693

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D045

October 26, 2001

INFORMATION NEEDED FOR CFL BREAK REVIEW, TAC MA9973

Reference: Firth, David J., "Transmittal of Final Report on the Evaluation of PSC 2-00 Relating to Core Flood Line Break with 2-Minute Operator Action Time," Letter to NRC from Manager, B&W Owners Group Services, Framatome ANP, FANP-01-988, April 2, 2001.

In assessing the Reference information, the significant remaining issue is the acceptability of a one minute reactor coolant pump (RCP) trip for design basis evaluation purposes. To complete our assessment, we require the following for those plants where credit is desired for a one minute RCP trip:

1. Provide applicable emergency operating procedures or describe any significant differences between the EOPs and the generic emergency procedures guidelines contained in Volumes 1 and 2 of Revision 09 of the Technical Bases Document. Describe operator training applicable to ensuring RCP trip within one minute of loss of subcooling margin. Assuming this is a simulator "critical task," include RCP trip time history from operator requalification testing.

2. Identify the information needed by the operators and describe how the information is provided visually and audibly. Address the safety "pedigree" of the applicable displays and describe alternates, including operator response, if the primary information sources are lost. Include photographs of the key visual information sources.

3. Identify RCP trip controls and describe how they operate. Include a description of the relative location of the controls and the applicable displays.

Question 1

Provide applicable emergency operating procedures or describe any significant differences between the EOPs and the generic emergency procedures guidelines contained in Volumes 1 and 2 of Revision 09 of the Technical Bases Document. Describe operator training applicable to ensuring RCP trip within one minute of loss of subcooling margin. Assuming this is a simulator "critical task," include RCP trip time history from operator requalification testing.

A. The following text is excerpted from TMI EOP ATP1210-10 Revision 41. This is 1 of 7 RULES which are committed to memory and performed by licensed operators upon recognition of a valid initiation condition. The condition for Rule 1 is subcooling margin indication < 25°F. Rule 1 has special precedence in the hierarchy for implementation. Rule 1 is performed as a priority over any other Emergency Procedure action. The hierarchy of emergency procedure actions is

SCM

1

described in OS-24 "Conduct of Operations during Abnormal or Emergency Events".

Rule 1

Loss of Subcooling Margin (SCM)

1. ***WHEN*** *SCM is less than 25 °F,*

   AND *reactor is shutdown,*

   THEN *immediately trip all operating RCPs.*

   - ***IF*** *RCPs were* ***NOT*** *tripped within 1 minute,*

     THEN *run at least one RCP per loop.*

2. *Ensure HPI and LPI are initiated.*

3. *Ensure EFW has auto started (REFER TO Guide 15).*

4. *Raise operable OTSG level to 75%-85% Operate Range (per Rule 4 – FWC).*

B. The actions in response to a loss of subcooling margin are "Critical Tasks". This sequence is practiced many times during the biennial training program. The time from Loss of Subcooling margin until Trip of the all of RCPs has been confirmed (during unannounced simulator exercises) to be typically between 15 to 30 seconds. Tripping the Reactor Coolant Pumps immediately upon loss of SCM has been a training program critical task for many years (since training critical task concept was programmatically instituted) and there have been no licensed operator failures of that task (i.e. once qualified no operator has ever failed this critical task). The delay time of each performance is not recorded, but the acceptance criteria has always been satisfied.

Question 2

Identify the information needed by the operators and describe how the information is provided visually and audibly. Address the safety "pedigree" of the applicable displays and describe alternates, including operator response, if the primary information sources are lost. Include photographs of the key visual information sources.

The operators are trained to use the SCM Indicators whenever there is RCS flow, in order to determine if SCM is adequate. (OS-24 Section 4.7).

The SCM indicators are digital indicators behind the main console on panel PCL. The indicators are visible from each Emergency response station within the control room (i.e. RO area at the console, CRS or STA standing behind the PPC console, or SM at desk on raised platform). The scripted response to a Reactor Trip (which should precede any LSCM) prescribes that the BackUp Reactor Operator immediately verify SCM while the Primary RO takes the immediate actions in response to a reactor trip. (OS-24 Section 4.3.5)

In addition, there is an RED overhead annunciator if SCM < 25°F (from either SCM instrument channel).

There are two independent "safety grade" instrument channels for RCS Subcooling Margin. The adequacy of the instrument channels and loop error was evaluated, reviewed by the NRC and documented in SECY-85-189.

Question 3

Identify RCP trip controls and describe how they operate. Include a description of the relative location of the controls and the applicable displays.

The reactor coolant pumps are operated via "pistol grip" type control switches located on the right side of the Main Console. The controls are immediately available from the reactor operators assigned duty station. The switches and breaker position indicating lights are visible from behind the PPC console, so that CRS, STA or SM may independently confirm that the action has been performed correctly.

CR-3 input to NRC RAI on PSC 2-00

1A. Provide applicable emergency operating procedures or describe any significant differences between the EOPs and the generic emergency procedures guidelines contained in Volumes 1 and 2 of Revision 09 of the Technical Bases Document.

The EOP TBD, Revision 9, Volume 1, Section III.B-1, Loss of SCM lists "Trip RCPs" as step 1.0 with a reference to Rule 1.0. Rule 1.0, step 1.1 states "Trip all RCPs Immediately" Rule 1, Note 1 provides contingency guidance to maintain one pump running in each loop if RCPs were not tripped within two minutes of losing SCM.

CR-3 procedure EOP-03, Step 2.1 (first immediate action step of the procedure) implements the requirements of the above referenced TBD sections.

On 12/12/2000 Framatome ANP issued TBD proposed change PC-2000-09 that included interim guidance requiring that the contingency action for failure to trip the RCPs be reduced from two minutes to one minute. CR-3 has reviewed their licensing basis and has determined that this change will require NRC review and approval. CR-3 plans to make this submittal in the first quarter 2002.

1B. Describe operator training applicable to ensuring RCP trip within one minute of loss of subcooling margin. Assuming this is a simulator "critical task," include RCP trip time history from operator requalification testing.

Tripping the RCPs upon recognition of a LOSCM is an EOP immediate action and a required operator memory item. Operators are trained to monitor RCS SCM using (1) the SPDS SCM Monitor, (2) the SPDS P/T display, (3) the Subcooling Margin Monitor slave units, and (4) when necessary by direct observation and plotting on error corrected curve.

Operators routinely respond to scenarios involving a LOSCM during simulator training and evaluation scenarios. These exercises involve a variety of event scenarios including large and small break loss of cool ant accidents as well as events involving LOSCM due to inadequate heat transfer.

Manual RCP trip upon recognition of a LOSCM is a "Time Critical Task" in all evaluation/examination scenarios. It is also carefully monitored during all training scenarios involving a LOSCM. CR-3 does not maintain a record of the time required to trip RCPs when the action was performed correctly within the allotted time window. Timing information would be recorded if the action is not performed correctly.

CR-3 training records over the last requalification training cycle reveal that there have been no failures associated with this time critical task. In response to this RAI, timings were recorded for six crews during the 2001 annual requalification examinations. For three of the crews the SCM monitor was functioning correctly. These three crews completed the action to trip RCPs within 10 seconds of losing SCM. The scenario for the other three crews included a failure of all SCM monitor alarm functions (see response to question 2 below for details on these functions). These three crews tripped RCPs in 10 seconds, 22 seconds, and 23 seconds respectively.

2. **Identify the information needed by the operators and describe how the information is provided visually and audibly. Address the safety "pedigree" of the applicable displays and describe alternates, including operator response, if the primary information sources are lost. Include photographs of the key visual information sources.**

Response

Primary indication of RCS SCM is provided by the Subcooling Margin Monitor (SCMM) which is part of the CR-3 Safety Parameter Display System (SPDS).

The SPDS/SCMM displays are located on the main control board directly above the reactor control panel (see Figure 1). The subcooling margin (or degrees superheat) presented on this display is calculated using inputs from RCS Loop A or B pressure, and either T_{hot} or $T_{incore,}$ as selected by switches on the SPDS control panel. Following a reactor trip the Subcooling Monitor automatically transfers to T_{incore} as the preferred temperature input.

Normally, the Subcooling Margin Monitor is displayed in the area at the top of the SPDS display. When not in alarm, the Subcooling Margin is displayed as black characters on a green background (see Figure 2).

If Incore temperature exceeds the SCM limit, the Subcooling Margin Monitors (1) sound a unique audible alarm, and (2) enlarge their displays to cover the entire display. The display color changes to a yellow background if the SCM limit is exceeded and a red background if the superheat limit is exceeded. The size of the display characters are increased as large as possible and a timer appears on the display (see Figure 3 – shown with <u>superheat limit</u> exceeded).

The timer on the SCM display starts at zero and counts up in seconds. This feature provideds positive indication of the elasped time since SCM limit was exceeded. The timer is reset whenever the Subcooling alarm resets.

When the "Acknowledge" button is pressed on the SPDS control panel, the audible alarm is silenced and the Subcooling Margin display reduces in size, and drops to the bottom of the SPDS display, retaining the color characteristics corresponding to the alarm limit exceeded. The selected SPDS graphic display

occupies the remaining available area above the Subcooling Margin display (see Figure 4 – shown with SCM limit exceeded).

If the Subcooling Margin alarm clears before it is acknowledged, it remains full screen but its colors change to black characters on a green background and the audible alarm continues to sound.

In addition to the primary SPDS displays, two Subcooling Margin Monitor Slave Display units are mounted on the PSA section of the main control board (see Figure 1 and 5). These units provide a duplicate display of the SPDS subcooling margin. The slave displays can be used to monitor SCM if the SPDS CRTs were to fail however they are dependent on their associated SDPS computer for intelligence.

Each slave unit is provided with a visual "heartbeat" feature (blinking light) to indicate an operable data link between the display and its respective SPDS computer. The display normally indicates SCM in steady, semi-bright characters. If the SCM limit is exceeded the characters go to full intensity and begin flashing.

If both SPDS displays and both Subcooling Monitor Slave Display units were to fail, subcooling margin would be determined by manually plotting the output of Reg Guide 1.97 pressure and temperature instruments on an instrument error corrected pressure/temperature curves (see Attachment 1) Reg Guide 1.97 instruments are clearly marked on the main control board and the instrument error corrected pressure/temperature curve is included in applicable EOPs

In 1999, CR-3 modified the SPDS subcooling margin display to serve as the primary indication of subcooling margin required by Improved Technical Specification 3.3.17, Table 3.3.17-1, Item 21 and the Bases. Details of the modification upgrade are discussed in our License Amendment Request #246, dated October 30, 1998 and approved in Amendment 174, dated April 20, 1999. The two redundant channels of SPDS are powered by independent emergency power sources and meet the requirement for Regulatory Guide 1.97 Type A, Category 1 instrumentation as described in LAR #246. For additional information concerning the "pedigree" of the SPDS system, see the attached excerpt from CR-3 Design Basis Document, Tab 5/11, Post Accident Monitoring Instrumentation, pages 10, 13, 16, 1722, and 23 and Tab 5/12, SDPS/RECALL System (Attachment 2).

3. **Identify RCP trip controls and describe how they operate. Include a description of the relative location of the controls and the applicable displays.**

Response

Reactor Coolant Pump control switches are located on the main control board below and to the left of the SPDS displays (see Figure 1 and 6). The RCPs are controlled by standard General Electric Type SBM pistol grip control switches. To stop an RCP motor the control switch is selected to the "STOP" position. If an individual RCP motor breaker were to fail to open on command, the pump can also be stopped from the main control board opening the feeder breaker to the electrical bus powering the affected RCP.

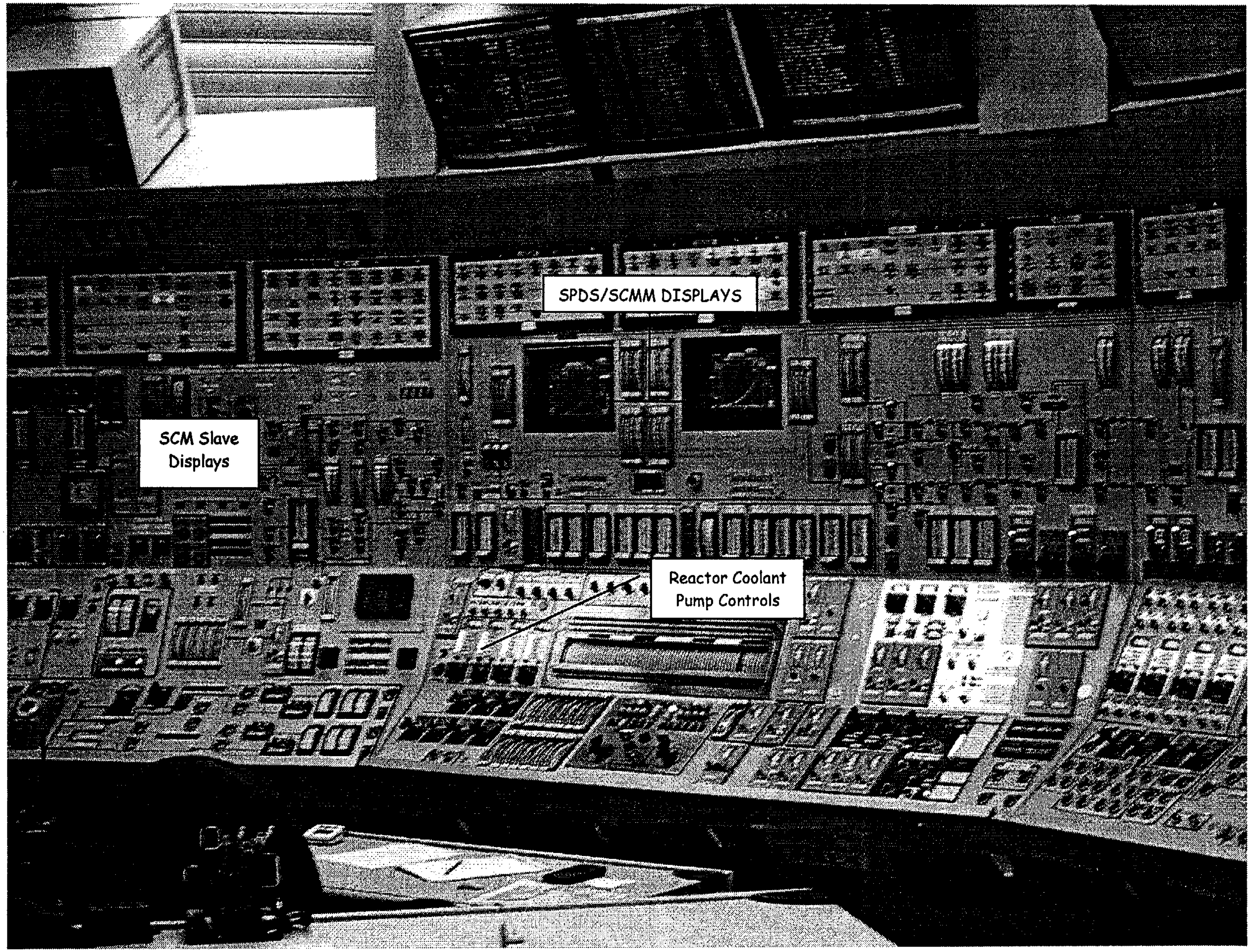
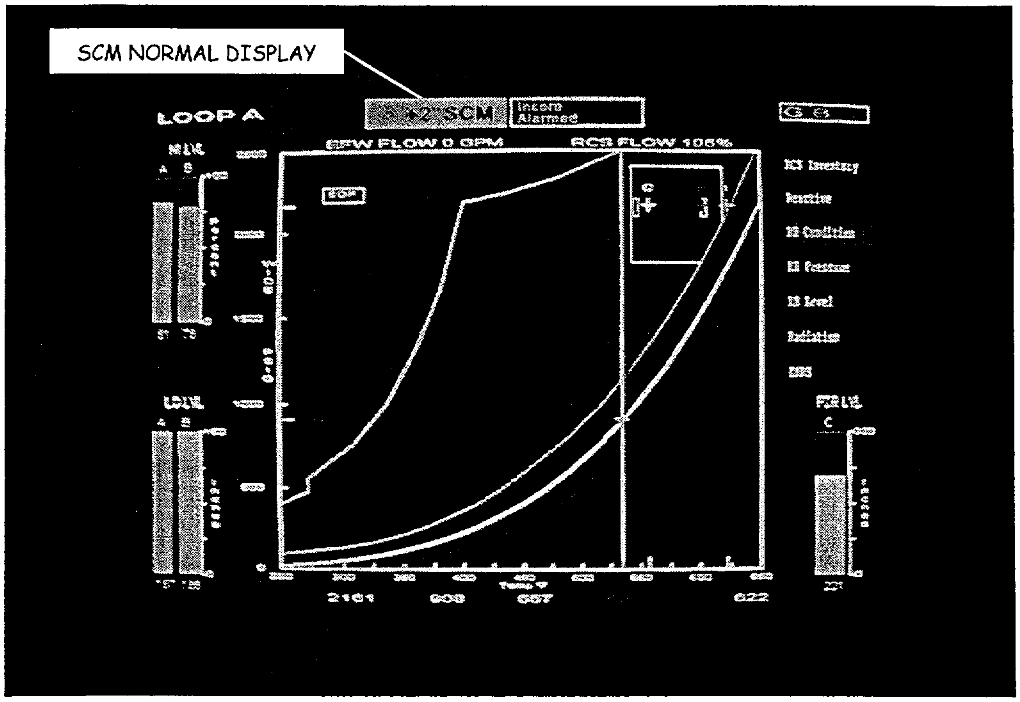

SPDS/SCMM DISPLAYS
SCM Slave Displays
Reactor Coolant Pump Controls

Figure 1



FIGU
RE 2

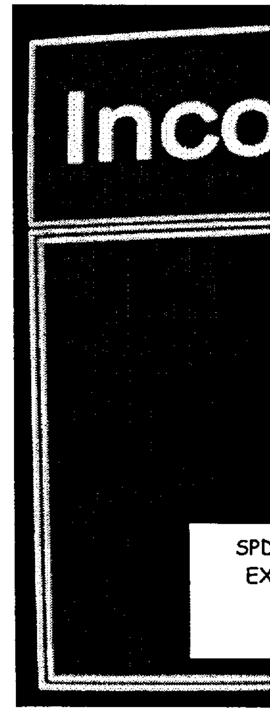


FIGURE 3

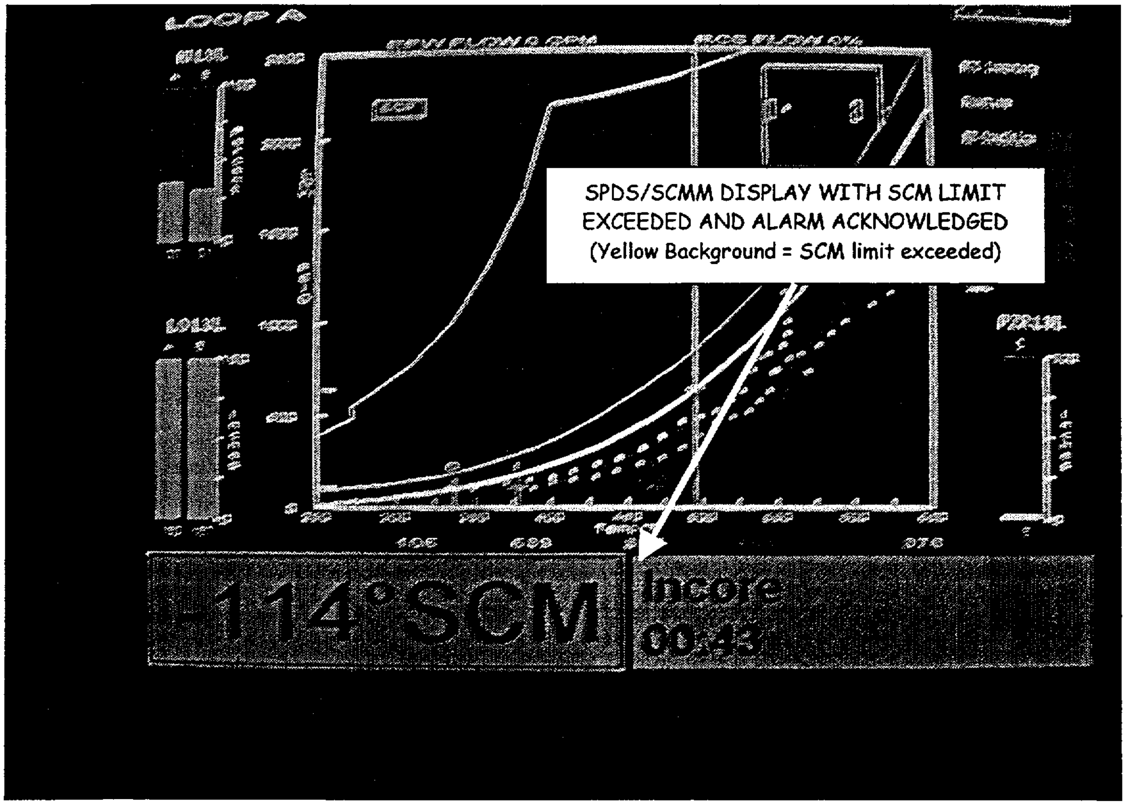


FIGURE 4

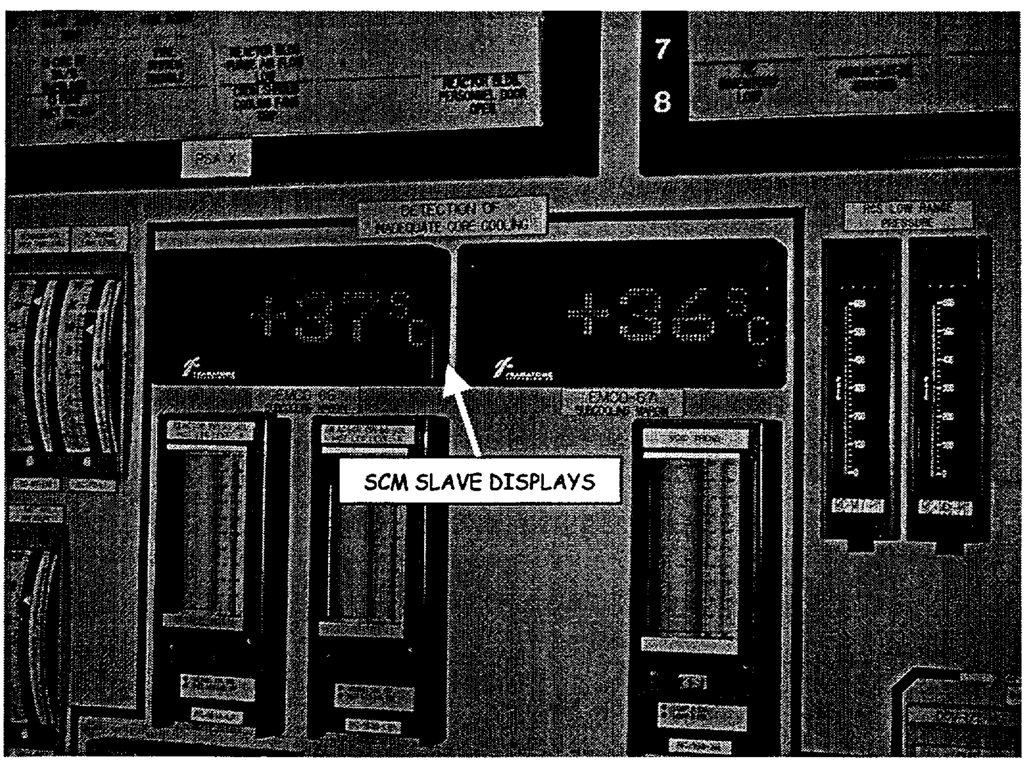


FIGURE 5

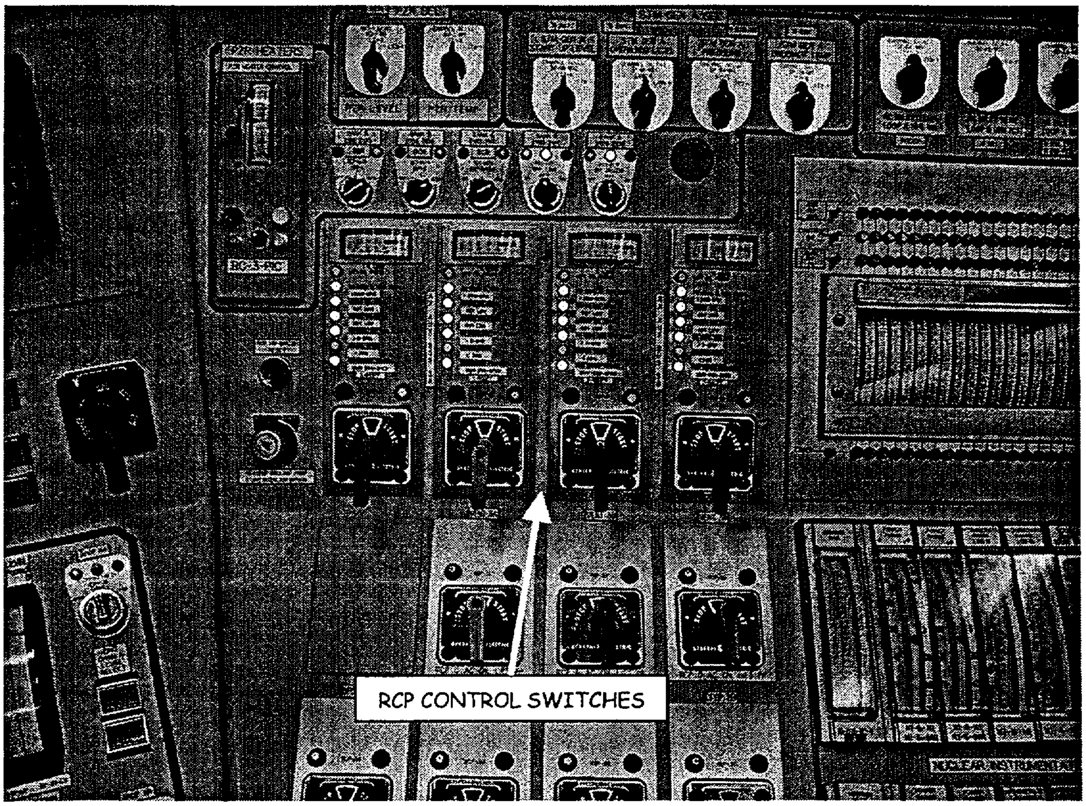


FIGURE 6

Florida Power
A Progress Energy Company

NUCLEAR OPERATIONS ENGINEERING
CRYSTAL RIVER UNIT 3

DESIGN BASIS DOCUMENT

FOR THE

POST ACCIDENT MONITORING INSTRUMENTATION

SYSTEM CODE: N/A
TAB: 5/11

ISSUE DATE: 09/5/86

	Revision 9	Revision 10	Revision 11	Revision 12	Revision 13	
Date	10-11-99	12-29-99	3-16-01	8-8-01	11-19-01	
Design Engineer	R.P.Schmiedel	R.P.Schmiedel	R.P.Schmiedel	R.P.Schmiedel	R.P.Schmiedel	
Verification Engineer	L.J.Santonastaso	P. A. Benyola	M. M. Loehr	Chris Sterner	L. McGowan	
Supervisor	G. E. Englert	L.J.Santonastaso	G. E. Englert	G. E. Englert	Ken Wilson	

REVISION LOG

Revision/Date	Description
0 / 9/05/86	Initial Issue
1 / 9/28/90	This revision is the result of modifications and changes made through Refuel 7. It revises all RG 1.97 variables to bring them into compliance with RG 1.97, Rev.3 Compliance Table to agree with previous commitments to the NRC contained in FPC letters 3F0388-18, dtd.3/21/88; 3F0189-11, dtd.1/25/89; 3F1289-12 dtd.12/15/89; 3F0190-06 dtd.1/19/90; 3F0890-06 dtd.8/2/90. It incorporates DBDTC #101 along with minor comments made during the review cycle. This revision also deletes DBDTC #66 and #83, as they were incorporated into DBDTC #101.
2 / 3/12/93	This revision incorporates DBDTC #087 and DBDTC #148 to pages 6, 31, and 91 as shown by the revision bars.
3 / 7/31/96	Incorporated DBD Temporary Changes #279, #303, #306, #324, #461 and #489 as identified by the change bars.
4 / 6/04/97	Incorporated Temporary Change #482 as indicated by change bars and added Source Document previously omitted from Temporary Changes #303 and #306.
5 / 8/01/97	Incorporated Temporary Changes #475 and #513 as indicated by the change bars.
6 / 4/17/98	Incorporated Temporary Changes #495, #592, #640, #593, #651, #624, #704, #745, and #788 as indicated by the change bars. Changes in TC #640 page 3 of 4 were shown to affect page 24; however these changes actually affected page 32. Changes in TC #788 were redundant and less detailed than those in TC #624 and TC #704 for page 64, therefore these changes were not incorporated.
7 / 6/15/99	Incorporated Temporary Changes #667, #867, #872, #879, #900 and #950 as indicated by the change bars.
8 / 8/18/99	Incorporated Temporary Changes #990 and #1050 as indicated by the change bars.
9 / 10/11/99	Incorporated Temporary Changes #929, 968, 986, and 1075 as indicated by the change bars.
10 / 12/29/99	Incorporate Temporary Changes #735, #992, #997, #1045, #1087, #1097 and #1117 as indicated by the change bars.
11 / 3/16/01	Incorporate Temporary Changes #1161 and #1215 as indicated by the change bars.
12 / 8/8/01	Incorporated Temporary Change #1235 as indicated by the change bar.

13 / 11-19-01 Incorporated Temporary Change #1251 as indicated by the change bars.

DESIGN BASIS INDEX
FOR
POST-ACCIDENT MONITORING INSTRUMENTATION

SECTION TITLE PAGE

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SECTION TITLE PAGE

SUMMARY SYSTEM DESCRIPTION

The post-accident monitoring instrumentation is comprised of instrumentation and displays to assess plant and environs conditions during and following an accident. Certain displays were added and/or upgraded in accordance with NUREG-0737 as TMI Lessons Learned Recommendations. Subsequently USNRC Regulatory Guide (RG) 1.97 of post-accident monitors was greatly expanded when supplement 1 to NUREG 0737 - "Requirements for Emergency Response Capability (Generic Letter 82-33)" was issued (12/17/82). The Crystal River Unit 3 degree of compliance is contained in the CR3 RG 1.97 Position Report submitted to the NRC August 21, 1984 on the basis of the events for which CR3 was licensed and revised reports were submitted to the NRC March 21, 1988 and December 5, 1990. Note, that this section of the Design Basis Document, (Section 5-11) replaces the Compliance Table previously submitted to the NRC.

Variables are grouped into five types depending on the importance of information as defined by RG 1.97, Rev. 3 and in addition each variable is assigned to one of three categories as a function of the safety importance of the measurement as follows:

Type A - Those variables that provide primary information* needed to permit the control room operator to take the specified manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety function for design basis accident events. They are plant specific and were selected on the basis of the CR3 Emergency Operating Procedures.

Type B - Those variables that provide information to indicate whether plant safety functions are being accomplished, defined as reactivity control, core cooling, primary coolant integrity, and containment integrity.

Type C - Those variables that indicate the potential for being breached or the actual breach of barriers to fission product release, including fuel cladding, primary coolant pressure boundary, and containment.

Type D - Those variables that provide information to indicate operation of individual safety systems and other systems important to safety.

Type E - Those variables that provide information for use in determining the magnitude of release of radioactive materials and for use in assessing such releases.

* Primary information is information that is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

Category 1 - These measurements are key variables with the most stringent requirements.

Category 2* - These variables have less stringent requirements and generally apply to the instruments designated for indicating system operating status.

Category 3 - Is intended to provide requirements to ensure that high quality off-the-shelf instruments are used for backup and diagnostic instrumentation.

* The general design approach for FPC for Category 2 variables is to make the sensor and first module of the string safety related and then take the buffered non-safety signal to the indicator.

The approach is not comprehensive as there are some exceptions, as documented on the variable sheets. The basis of this position is explained in B. ENVIRONMENTAL QUALIFICATION.

Areas of qualification including range, environmental qualification, seismic qualification, quality assurance, redundancy, power source, display, schedule, position, source and reason are listed and explained below.

A. RANGE

The ranges listed in the Compliance Table are the actual measurement range of the variable at CR3. If the range varies from that stated in the Regulatory Guide justification is supplied for the existing range. In some instances, the Regulatory Guide states the range in terms of a percentage of the design. In these cases, the design basis is listed next to the range in parenthesis.

B. ENVIRONMENTAL QUALIFICATION

A response of "Yes" on the Compliance Table indicates that the currently installed equipment located in a "Harsh" environment meets the requirements of IE Bulletin 79-01B and 10CFR50.49. Equipment located in "Mild" environments are not part of the EQ Program. This determination was based on either having actual environmental qualification documentation available or documentation on similar equipment available.

For Category 2 variables, FPC considers existing installed instrumentation located in a mild environment to be adequate for Regulatory Guide 1.97 Category 2 variables. FPC also considers portions of the Non-nuclear Instrumentation (NNI) adequate for Category 2 variables and has the following position:

> For strings, which include hardware located in a harsh environment, portions in the harsh environment (sensors, cabling, terminations) should be qualified for the accident temperature, pressure, humidity, radiation and chemical environment. Hardware located in a mild environment (cabling, terminations, processing modules, power supplies, indicators and recorders) is adequate as currently installed.

The basis for this position is as follows:

> The Category 2 qualification requirements of Regulatory Guide 1.97, Rev. 3, include no specific provision for seismic qualification. We interpret this to mean that environmental qualification only is required. Since 10CFR50.49 does not require environmental qualification for equipment located in a mild environment, only those components listed in a harsh environment need be qualified.
>
> The currently installed NNI equipment was not supplied as safety related equipment but is comparable in quality and reliability to existing safety related equipment. In fact, some of the NNI electronic modules are identical to those qualified and supplied for these safety related systems. Operating experience with the NNI indicates that this instrumentation can reasonably be expected to be operable for accident monitoring.
>
> Category 2 instrumentation is not required to be seismically qualified, redundant, physically and electrically separated nor powered from a 1E source. The existing NNI

hardware located in a mild environment is consistent with the Category 2 criteria and no substantial improvement in reliability or safety would be expected if this equipment were replaced with new, qualified hardware.

Other responses are self-explanatory.

C. SEISMIC QUALIFICATION

A response of "Yes" on the Compliance Table indicates that the entire instrument string is seismically qualified in accordance with Regulatory Guide 1.100. Other responses are self-explanatory.

D. QUALITY ASSURANCE

A response of "Yes" on the Compliance Table indicates that Quality assurance requirements meeting CR3's licensing commitments as documented in the FSAR Section 1.6 were applied to at least the safety related portions of the instrument string. All other responses are self-explanatory.

E. REDUNDANCY

A response of "Yes" indicates that redundant channels are available up to and including any isolation device and that the channels are both electrically independent and physically separate from each other, in accordance with IEEE Standard 279-1971, and meet single failure criteria. All other responses are self-explanatory.

F. POWER SOURCE

The power source for the instrument string listed n the Compliance Table is in compliance with the Regulatory Guide requirements unless otherwise noted.

G. DISPLAY

Under this heading on the Compliance Table is how the variable is indicated and/or recorded in the Control Room (CR), EFIC Room, etc.

If the variable is available on demand in the Technical Support Center (TSC) or the Emergency Operating Facility (EOF) it will be so stated.

H. SCHEDULE

This area indicates when the upgrades (if required) will be complete.

I. SOURCE

This area indicates the source documents, which can be found on page 91, for each of the variables.

J. REASON

In this area will be Florida Power Corporation's position on a particular variable which will include any justifications which are required along with any comments or clarifying remarks which may be needed.

If the justification presented is justification developed by the Babcock & Wilcox Owners Group (BWOG) Regulatory Guide 1.97 Task Force, it will be so stated.

DESIGN BASIS FOR
POST-ACCIDENT MONITORING INSTRUMENTATION

PARAMETER:

VARIABLE: NEUTRON FLUX
TAG NO.: NI-14-NI1, NI-15-NI1, NI-15-NIR
REF DWG: 205-042, NI-01

Type and Category - A, B, 1

Range - 10^{-8} to 100%, (SR, IR, PR)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 channels - PR 2 channels - SR 2 channels - IR

Power Source - 1E

Display - Indicated and Recorded in CR
On Demand in EOF & TSC

SOURCE:

0, 5, 6, 7, 18, 21, 42

REASON:

Neutron flux is the measure of reactor power required to monitor reactivity control of the ICS and RPS.

PARAMETER:

VARIABLE: CONTROL ROD POSITION
TAG NO.: DR-1-KI thru DR-69-KI (Panel Meters)
DR-70-KI thru DR-73-KI (Group Average Meters)
REF DWG: 210-074

Type and Category - B, 3

Range - 0 - 100%, Full-in/Full-out Lights, Average Group Position

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A - 2 Channels (API and RPI)

Power Source – Reg. Inst. Bus VBDP-1 & VBDP-2

Display - Indicated in CR
Average Group Position on Demand in TSC & EOF

SOURCE:

0, 5, 33, 34, 42

REASON:

Control rod position provides backup information that reactivity control has been accomplished by the ICS and RPS.

NOTE:

DR-30-KI (Panel Meter) and DR-30-KT (PI Tube) no longer exist. Removed to provide for RCITS (Reactor Coolant Inventory Tracking System)

PARAMETER:

VARIABLE:	LPI Pump Status (Running)	
TAG NO.:	For DHP-1A:	CB-ESFA-LX3 or CB-ESFA-HU;
	For DHP-1B:	CB-ESFB-LX3 or CB-ESFB-HU
REF. DWG:	208-021, Sheets DH-01 and DH-02	

Type and Category – A, 1

Range – ON / OFF

Environmental Qualification – N/A (Mild Environment)

Seismic Qualification – Yes

Quality Assurance – Yes

Redundancy – Yes, 2 trains

Power Source – 1E

Display – Indicated in the CR, recorded on the RECALL Computer

SOURCE:

54, 55

REASON:

Operator verifies LPI Pump Run Status during several evolutions. During LOCA, the LPI pump is manually aligned to provide suction to the associated HPI pump, in a configuration called LPI/HPI piggyback. These actions are necessary whenever the inventory in the Borated Water Storage Tank (BWST) is nearing depletion and the size of a LOCA is such that RCS pressure remains higher than shutoff head of the LPI pump. In addition, LPI/HPI piggyback operation is required whenever single failures occur resulting in only one train of LPI being available. Since the HPI pump cannot take suction directly from the Reactor Building (RB) sump, the only way to continue HPI injection is to place the systems into the LPI/HPI piggyback mode. Operator has to verify that the LPI pump is operating prior to opening the applicable "Piggyback" valve (DHV-11 and DHV-12) between the LPI pump discharge and the associated HPI pump suction.

LPI Pump Run Status is a RG 1.97 Type A variable because it is used by the Operator to determine if LPI/HPI piggyback operation can be implemented as required to mitigate LOCAs and maintain long-term emergency core cooling.

PARAMETER:

VARIABLE: DHV-42 and DHV-43 Valve Position (Open)
TAG NO.: CB-ESFA-KN3, CB-ESFB-KN3
REF. DWG: 208-021, Sheets DH-14 and DH-15

Type and Category- A, 1

Range – OPEN / CLOSED

Environmental Qualification – Yes

Seismic Qualification – Yes

Quality Assurance - Yes

Redundancy – Yes, 2 trains

Power Source – Yes, 480 VAC MCCs, Diesel-backed power supplies

Display – Indicated in the CR, not recorded

SOURCE:

54, 55

REASON:

The Operator verifies that valves DHV-42 and DHV-43 are open to ensure that suction has been established from the RB Sump for the LPI Pumps and Building Spray (BS) pumps prior to isolating suction from the BWST using valves DHV-34 and DHV-35. Verification that valves DHV-42 and DHV-43 are open is necessary to preserve the operability of the LPI and BS pumps, and possibly HPI pump during LPI/HPI piggyback operation, by allowing isolation of pump suction from the BWST and preventing possible cavitation of the operating pumps.

DHV-42 and DHV-43 Open position is RG 1.97 Type A variable because it is used by the Operator to determine if completion of transfer from BWST to RB sump can be accomplished as required to mitigate LOCAs and to maintain long-term emergency core cooling and containment cooling.

Exception to the recording requirement applied for to the NRC in License Amendment Request 234.

PARAMETER:

VARIABLE: HPI Pump Status (Running)
TAG NO.: For MUP-1A: CB-ESFA-MF7 or CB-ESFA-AH; For MUP-1B on the "A" train: CB-ESFA-MN7 or CB-ESFA-AJ; For MUP-1C: CB-ESFB-MF7, or CB-ESFB-AH; For MUP-1B on the "B" train: CB-ESFB-MV7 or CB-ESFB-AJ
REF. DWG: 208-041, Sheets MU-01, MU-02, MU-03, MU-04

Type and Category – A, 1

Range – ON / OFF

Environmental Qualification – N/A (Mild Environment)

Seismic Qualification – Yes

Quality Assurance – Yes

Redundancy – Yes, 3 pumps and 2 trains. MUP-1B can be aligned to either the A or B train

Power Source – 1E

Display – Indicated in the CR, Recorded on the RECALL Computer

SOURCE:

54, 55

REASON:

During emergency operations, the two ES selected HPI pumps are designed to automatically start. The Operator has to verify the operating HPI Pump(s) Status during several evolutions. During LOCAs, the HPI pump run status is necessary to comply with required EOP actions for opening of the HPI pump recirculation valves to protect the necessary HPI pump(s) from damage due to low flow conditions. The EOPs specify different minimum pump flow rates for opening the HPI pump recirculation valves depending upon the number of HPI pumps in operation.

HPI Pump Run Status is a RG 1.97 Type A variable because it is used by the Operator to ensure protection of the operating HPI pump(s) from damage due to low flow conditions as required to mitigate LOCAs and maintain long-term emergency core cooling.

PARAMETER:

VARIABLE: RCS SOLUBLE BORON CONTENT
TAG NO.: CA-56-CE, CA-56-CI
REF DWG: 302-700, 209-010

Type and Category - B, 3

Range - 0 - 6000 ppm

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - 1E/DG

Display - Pass Analyzer Panel

SOURCE:

0, 5, 10, 11, 12, 42

REASON:

The Boronmeter reading on the Pass Analyzer Panel (Local Indication) is sufficient to meet the intent of Regulatory Guide 1.97, Rev. 03. This is based on the fact that the loss of negative reactivity due to xenon decay is sufficiently slow that the Control Room operator need not know instantaneously or continuously what the boron concentration is in the RCS. Also, Section II.B.3 of NUREG-0737 requires that capability exists to sample and analyze the reactor coolant in a post-accident environment. The PASS Upgrade Project installed the Boronmeter in 1995. This is an NRC accepted exception, per NRC SER Docket No. 50-302, Conformance to Regulatory Guide 1.97, dated 6/16/87.

PARAMETER:

VARIABLE: RCS HOT LEG WATER TEMPERATURE
TAG NO.: RC-4A-TI4-1, RC-4B-TIR1
REF DWG: 205-047, RC-04, RC-10, RC-12A, RC-13A

Type and Category - A, B, 1

Range - 120 - 920°F

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E/DG

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 10, 11, 13, 19, 42

REASON:

RCS Hot Leg Water Temperature is a key variable required to monitor the core cooling safety function, to verify natural circulation along with core exit temperatures, and to verify primary to secondary loop coupling along with steam generator pressure.

RCS Hot Leg Temperature not required below 280°F. Plant in cold shutdown below 200°F. RCS Cold Leg Temperature range extends down to 50°F. This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

RCS Hot Leg temperature is a Type A variable because it is used by the operator to determine subcooling margin (along with RCS pressure) if SPDS is not available. When subcooling margin is lost, the operator must trip RC pumps.

PARAMETER:

VARIABLE: RCS COLD LEG WATER TEMPERATURE
TAG NO.: RC-5A-TI1, RC-5B-TI1
REF DWG: D8034033 Sht. 3A

Type and Category - B, 3

Range - 50°F - 650°F (Ind)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A - 2 Channels

Power Source - 1E/DG

Display - Indicated in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 10, 11, 42

REASON:

Reg. Guide 1.97 lists Cold Leg Water Temperature as a Category 1 (key) variable and Core Exit Temperature as a Category 3 (backup) variable for the core cooling function. Cold Leg Temperature indication may not in all cases provide valid information on the status of core cooling. Since it is located in the RCS loops and not the reactor vessel, there must be either forced or natural circulation flow through the steam generators for indications to be representative of actual core conditions. Also, due to the proximity of the cold leg RTDs to the HPI nozzles, HPI flow may significantly affect the cold leg temperature indication particularly in the absence of forced RCS flow. Incore temperature monitors provide a more direct indication of core cooling independent of whether or not there exists coolant flow through the loops. RCS Cold Leg Water Temperature is a backup to RCS Hot Leg and Core Exit Temperatures.

The key variables for monitoring the core cooling plant safety function are RCS Hot Leg Water Temperature, Core Exit Temperature, and Steam Generator Pressure (see Discussion Section for RCS Hot Leg Water Temperature). RCS Cold Leg Water Temperature is a backup temperature monitor to the RCS Hot Leg Water Temperature and Core Exit Temperature.

For these reasons, core exit temperature and RCS Hot Leg are the key variables for monitoring core cooling and are qualified to Category 1 requirements while RCS Cold Leg Temperature serves as a backup variable and is qualified to Category 3 requirements accordingly.

The CR3 range of 50° to 650°F is based on providing the capability of the RCS Cold Leg Water Temperature instrumentation to measure a value greater than the saturation temperature for the steam generators, which is approximately 500°F (based on 1050 psig design pressure). 650°F for the high end of the range provides 15% excess measurement capability and is approximately 110% of the design temperature of 600°F. The low end of the range, 50°F, allows for measurement of the variable during conditions where the DHRS or LPI system is not in use or available and the steam generators are removing decay heat.

The range and category of these instruments are an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: RCS PRESSURE

TAG NO.	REF. DWG.	Type and Category	Range
RC-158-PI2	205-047, RC-02	A, B, C, 1	0 – 3000 psig
RC-158-PIR	205-047, RC-02	A, B, C, 1	0 – 3000 psig
RC-159-PI2	205-047, RC-02	A, B, C, 1	0 – 3000 psig
RC-147-PI1	205-047, RC-03	B, 1	0 – 600 psig
RC-148-PI1	205-047, RC-03	B, 1	0 – 600 psig

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E/DG

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 12, 13, 19, 52

REASON:

RCS pressure is a key variable required to monitor reactor shutdown in event of a reactor coolant upset and to monitor reactor coolant integrity and core cooling capability.

RCS Pressure is a Type A variable for the following reasons:

1.) In an inadequate core cooling scenario, the operator opens the PORV at 2400 psig and recloses it when pressure is reduced to >100 psig than the next higher ICC region.

2.) In an inadequate subcooling margin or station blackout scenario, the PORV is opened at 2400 psig and closed at 1600 psig.

3.) The operator uses RCS pressure (along with RCS Hot Leg temperature) to determine subcooling margin if SPDS is not available. When subcooling margin is lost, the operator must trip RC pumps.

PARAMETER:

VARIABLE: REACTOR COOLANT INVENTORY
TAG NO.: RC-163A-LR1, RC-163B-LR1, RC-164A-LR1, RC-164B-LR1, RC-169-XR[1]
REF DWG: 205-047, RC-12, RC-12A, RC-13, RC-13A

Type and Category - B, 1 - RV. LVL and Hot Leg LVL
B, 2 - Void Fraction[2]
B, 3 - RCS Cold Leg Temperature Feeding Void Fraction[3]

Range - Bottom of Hot Leg to Top of Hot Leg
Bottom of Hot Leg to Top of Vessel

Environmental Qualification - Yes

Seismic Qualification - Yes (Category 1 Variable Only)

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E (Category 1 Variable Only)

Display - Indicated and Recorded in CR

SOURCE:

0, 5, 31, 42, 61

REASON:

RCS hot leg level is required to monitor, along with RC pump monitors, that no voids exist in the reactor coolant system.

Per Deficiency Report (DR) 99-0044, the following components would suffer consequential failure as a result of a Core Flood (West) line rupture:

- Instrument tubing for hot leg level indication instrument RC-163A-LT1
- Conduit RCR-315 for hot leg and reactor vessel level indication instruments RC-163A-TE1 and RC-164A-TE1, respectively.

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1 This tag number is associated with but isolated from the coolant inventory variable and is, therefore, identified as a Category 2 variable.

2 When the pumps are running, the void fraction indicators provide inventory level trend measurement.

3 RCS Cold Leg temperature is an input to void fraction

Justification for the acceptability of this failure has been provided in SA/USQD No. 99-0255 and included in the Topical Design Basis Document for High Energy Line Breaks Inside Containment (Reference 61).

PARAMETER:

VARIABLE: DEGREES OF SUBCOOLING
TAG NO.: EMCO-38, EMCO-39 (SPDS Monitors), RC-147-PT Recall Point 243, RC-3A-PT3 Recall Point 4, RC-4A-TE1 Recall Point 17, RC-4A-TE4 Recall Point 239, RC-148-PT Recall Point 40, RC-3B-PT3 Recall Point 5, RC-4B-TE1 Recall Point 18, RC-4B-TE4 Recall Point 240
REF DWG: SPDS Description Document, I-84-0003

Type and Category - A, B, 1

Range - 669°F Subcooled to 2288°F Superheated

Environmental Qualification - Yes

Seismic Qualification – Partial*

Quality Assurance - Yes

Redundancy – 2 Independent SPDS Channels*

Power Source - UPS/DG

Display - Indicated in CR

*The Subcooling Margin Monitor provided by the SPDS partially complies with Reg. Guide 1.97, type A, category 1 variable. See references 62 and 63.

SOURCE:

0, 5, 42, 45, 46, 47, 62, 63

REASON:

Subcooling margin in °F is displayed on both of the SPDS displays, EMCO-38 and EMCO-39, at the top center of the screen for all SPDS displays. The SPDS displays margin to saturation for each loop. The saturation temperature is determined from either a low range or wide range pressure instrument. The subcooling margin displayed is the difference between the saturation temperature and either T_{hot} (RTD's) or the highest of 8 qualified core exit thermocouple temperatures.

This variable is a Type A because the operator uses a loss of Subcooling Margin to trip the RC Pumps within 2 minutes of the loss of Subcooling Margin.

NRC issued Amendment Number 162 on December 22, 1997, approving revisions to the CR-3 improved Technical specification (TS), submitted by TSCRN 209, Revision 1, related to the post-accident monitoring instrumentation. TSCRN-209, Revision 1, states that the DEGREES OF SUBCOOLING has been reclassified as a Regulatory Guide 1.97 Type A, Category 1 variable with exceptions since the instrumentation used to measure the DEGREES OF SUBCOOLING does not meet all of the recommended criteria of a RG 1.97 Category 1 instrument.

Subsequent to TSCRN-209, License Amendment Request #246, Subcooling Margin Monitoring Using SPDS, dated 10/30/98, was submitted to the NRC. This LAR proposed to upgrade the SPDS to meet most but not all of the Reg. Guide 1.97 requirements as implemented by MAR 96-11-03-01. The NRC letter, Staff Evaluation and Issuance of Amendment Regarding Subcooling Margin Monitor Using SPDS (TAC No. MA4147), dated 4/20/99 (License Amendment No. 174) approved this approach.

The primary display of subcooling margin is the SPDS. A backup display of subcooling margin is obtained by manually plotting pressure and temperature data from category 1 Reg. Guide 1.97 RC pressure and temperature instruments on instrument error corrected figures.

PARAMETER:

VARIABLE: CONTAINMENT SUMP WATER LEVEL (SUMP)
TAG NO.: WD-301-LI, WD-302-LI, WD-301-LR, WD-302-LR
REF DWG: 205-060, WD-01

Type and Category - B, C, 2

Range - 0 - 10 ft.

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A - 2 Channels

Power Source - UPS/DG with 1E Standby

Display - Indicated in CR
On Demand in EOF and TSC
REC in EFIC Room

SOURCE:

0, 5, 22, 42

REASON:

Containment sump level is an important method of leak detection inside containment, including the reactor coolant system.

Location of the recorder in EFIC Room satisfies NUREG-0737, Item II.F.1.5. as documented in the Safety Evaluation Report, Docket No. 50-302, dated January 13, 1984.

PARAMETER:

VARIABLE: CONTAINMENT SUMP WATER LEVEL (FLOOD)
TAG NO.: WD-303-LI, WD-304-LI, WD-303-LR, WD-304-LR
REF DWG: 205-060, WD-03

Type and Category - B, C, 1

Range - 0 - 10 ft. (Above Sump)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E

Display - Indicated in CR
Recorded in EFIC Room
On Demand in EOF & TSC

SOURCE:

0, 4, 5, 22, 42

REASON:

Containment level is an important method of detection and inventory of a major fluid loss, including a LOCA.

Location of the recorder in the EFIC Room satisfies NUREG-0737, Item II.F.1.5. as documented in the Safety Evaluation Report, Docket No. 50-302, dated January 13, 1984.

PARAMETER:

VARIABLE: CONTAINMENT ISOLATION VALVES POSITION (MANUAL)
TAG NO.: N/A
REF DWG: N/A

Type and Category - B, 1

Range - N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

0, 5, 10, 11, 23, 42

REASON:

Containment isolation valve position is required to ensure containment integrity in event of a LOCA.

Locked closed manual valves do not require position indication.

PARAMETER:

VARIABLE: CONTAINMENT ISOLATION VALVES POSITION (AUTOMATIC)
TAG NO.: SEE ES LIGHT MATRIX
REF DWG: 201-162

Type and Category - B, 1

Range - Open/Closed Lights (via Light Matrix)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Redundant indication per valve not intended by RG 1.97, since CR3 has redundant isolation barriers for all fluid penetrations.

Power Source - 1E

Display - Indicated in CR

SOURCE:

0, 5, 10, 11, 13, 19, 42

REASON:

Containment isolation valve position is required to ensure containment integrity in event of a LOCA.

The redundancy exception is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: CORE EXIT TEMPERATURE (SAFETY RELATED)
TAG NO.: RC-171-TR (displays IM-3L, 5G, 6C, 6O, 9E, 9H, 10O, 13G-TE)
RC-172-TR (displays IM-2G, 4N, 6L, 7F, 10C, 10M, 11G, 13L-TE)
REF DWG: 205-047, RC-07, RC-08

Type and Category - A, C, 1

Range - 0° - 2500°F

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes

Power Source - UPS/DG

Display - 16 CETs Recorded in CR

SOURCE:

0, 5, 31, 42, 62, 63

REASON:

Core exit temperatures along with RCS Hot Leg Temperature to verify natural circulation of reactor coolant, and to detect potential breach of fuel cladding.

Ref: The NRC's evaluation of CR3 is response to NUREG-0737, Item II.F.2, Docket No. 50-302, dated 9/6/83.

There are 52 Core Exit Thermocouples (CETs) recorded on demand in the Control Room over a range of 0-2500°F. (Twelve (12) are also recorded on demand in the TSC and EOF over a range of 0-2000°F.)

There are 16 safety related temperature measurements from 16 CETs - 4 from each core quadrant. The system is part of the ICC detection system and is Class 1E. Each of the 16 Core Exit Temperature measurements is continuously recorded in the CR. The 16 strings are divided between two separate recorders over a range of 0-2500°F. The 16 safety related CET's are divided into two separate channels of 8 thermocouples each. Each channel has two CET's per core quadrant and is displayed on a separate recorder.

Core Exit Temperature is a Type A variable because it is used by the operators to determine subcooling margin if SPDS and RCS Hot Leg temperature are not available.

PARAMETER:

VARIABLE: CORE EXIT TEMPERATURE (NON SAFETY RELATED)
TAG NO.: IM-1H, 2L, 3F, 3M, 4E, 5D, 5H, 5K, 5O, 6G, 6P, 7B, 7E, 7M, 7R, 8B, 8F, 8H, 8N, 9C, 9G, 9M, 9N, 10D, 10R, 11E, 11K, 11L, 12F, 12K, 12O, 13C, 13H, 13F, 14D, 14M, -TE
REF DWG: N/A

Type and Category - B, C, 3

Range - 0° - 2500°F

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - 52 CETs Indicated in CR (16 safety related + 36 non safety related)
16 Recorded On Demand in CR (16 safety related)
12 On Demand in TSC & EOF

SOURCE:

0, 5, 31, 42, 62, 63

REASON:

Core exit temperatures along with RCS Hot Leg Temperature to verify natural circulation of reactor coolant, and to detect potential breach of fuel cladding.

Ref: The NRC's evaluation of CR3 is response to NUREG-0737, Item II.F.2, Docket No. 50-302, dated 9/6/83.

There are 52 Core Exit Thermocouples (CETs) recorded on demand in the Control Room over a range of 0-2500°F. (Twelve (12) are also recorded on demand in the TSC and EOF over a range of 0-2000°F.)

There are 16 safety related temperature measurements from 16 CETs - 4 from each core quadrant. The system is part of the ICC detection system and is Class 1E. Each of the 16 Core Exit Temperature measurements is continuously recorded in the CR on two separate recorders over a range of 0-2500°F.

The 36 non safety related CET's displayed on the plant computer are not class 1E but are energized from a battery backed, high-reliability uninterruptible power supply.

PARAMETER:

VARIABLE: RCS RADIOACTIVITY CONCENTRATION
TAG NO.: N/A
REF DWG: N/A

Type and Category - C, 3

Range – N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source – N/A

Display – N/A

SOURCE:

0, 5, 10, 11, 12, 42

REASON:

The following position is a justification developed by the BWOG Reg. Guide 1.97 Task Force.

Currently, no instrumentation exists to adequately measure this variable on line. Existing instrumentation, letdown line radiation monitors can be used to provide indication of fuel failure during normal operation. However, since the letdown line is isolated during serious accidents requiring containment isolation, it will not be available for long term measurement. Section II.B.3 of NUREG-0737 requires that capability exist at each plant to sample the RCS to assess the magnitude of fuel failures during post-accident conditions. As such, this measurement should be the primary determinant of fuel failure during normal operation and post-accident. The letdown line radiation monitor should be used as the initiator for sampling during normal operation because state-of-the-art equipment is unavailable and the primary means of monitoring this variable must therefore be by sampling and analysis.

This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

The sample system is used to meet the requirements of NUREG-0737 Section 11.B.3 RM-L1 is not used for RG 1.97 requirements since it does not have the sensitivity; however, it is used for trending.

PARAMETER:

VARIABLE: CONTAINMENT HYDROGEN CONCENTRATION
TAG NO.: WS-10-CR, WS-11-CR
REF DWG: 205-062, WS-01, WS-02

Type and Category - C, E, 1, 3

Range - 0 - 10%

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E

Display - Indicated and Recorded in EFIC Room

SOURCE:

0, 5, 22, 42

REASON:

Containment hydrogen monitoring is a key variable used to detect a potential breach of containment resulting from fuel failure.

Location of indicators and recorders satisfies NUREG-0737, Item II.F.1.6. This is an NRC accepted exception, per NRC SER docket No. 50-302, NUREG 0737 Items, dated 1/13/84.

PARAMETER:

VARIABLE: CONTAINMENT PRESSURE
TAG NO.: BS-16-PI, BS-17-PI, BS-90-PI, BS-91-PI, BS-90-PR, BS-91-PR
REF DWG: 205-009, BS-01 and BS-02

Type and Category - B, C, 1

Range - -10 - 70 psig (BS-16, 17-PI)
0 - 200 psig (BS-90, 91-PI)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E

Display - Indicated in CR
Recorded in EFIC Room
On Demand in TSC & EOF

SOURCE:

0, 5, 22, 42

REASON:

Containment pressure is a key measurement used for detection of a LOCA, verification of ESFAS mitigation, or detection of a potential breach of containment.

Recorder location meets NUREG-0737, Item II.F.1.4. This is an NRC accepted exception, per NRC SER Docket No. 50-302, NUREG 0737 Items, dated 1/13/84.

PARAMETER:

VARIABLE: DHHE OUTLET TEMPERATURE
TAG NO.: DH-2-TI1, DH-2-TI2
REF DWG: 205-021, DH-04 and DH-05

Type and Category - D, 2

Range - 0 - 300°F

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A - 2 Channels

Power Source - UPS with 1E Standby

Display - Indicated in CR

SOURCE:

0, 5, 10, 11, 13, 20, 42

REASON:

DHHE outlet temperature is used to monitor operation of the LPI system after a LOCA.

RTD is mounted in LPI piping.

Range covers all anticipated requirements. Design temperature of the Decay Heat System and Heat Exchanger for CR3 is 300°F.

This indicated range is a NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: CORE FLOOD TANK LEVEL
TAG NO.: CF-2-LI1, CF-2-LI2, CF-2-LI3, CF-2-LI4
REF DWG: D8034038

Type and Category - D, 3

Range – 1 to 14 ft. above tank lower instrument tap (approximately 3 ft. above tank base)

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A - 2 Channels

Power Source - UPS/DG

Display - Indicated in CR

SOURCE:

0, 5, 10, 11, 16, 36, 42, 56

REASON:

CF tank level is required to monitor safety injection in event of a LOCA.

Category 3 is an NRC accepted exception, per NRC SER Docket No. 50-302, Accumulator Pressure and Volume Instrumentation Relaxation of Regulatory Guide 1.97 Environmental Qualification Requirements, dated 12/16/93.

PARAMETER:

VARIABLE: CORE FLOOD TANK PRESSURE
TAG NO.: CF-1-PI1, CF-1-PI2, CF-1-PI3, CF-1-PI4
REF DWG: D8034038

Type and Category - D, 3

Range - 0 - 800 psig

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A - 2 Channels

Power Source - UPS/DG

Display - Indicated in CR

SOURCE:

0, 5, 10, 11, 12, 42, 56

REASON:

Core Flood Tank Pressure is a key variable for pre-accident status to assure that this passive safety system is prepared to discharge into the RCS in the event of a LOCA. This pressure indication provides no essential information for operator action during or following an accident. The key variable necessary to determine whether the Core Flood Tanks have fulfilled their safety function is Core Flood Tank Level. Therefore, Core Flood Tank Pressure is a backup type variable and has been classified as a Category 3 instrument accordingly.

This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: CORE FLOOD TANK ISOLATION VALVE POSITION
TAG NO.: CFV-5, CFV-6
REF DWG: 302-702

Type and Category - D, 3

Range - Closed/Open Lights

Environmental Qualification - No

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A - 2 Channels

Power Source - UPS/DG

Display - Indicated in CR

SOURCE:

0, 3, 5, 13, 20, 36, 42

REASON:

CF Tank Isolation Valve position is required to monitor that valve operational status is correct.

PARAMETER:

VARIABLE: BORIC ACID CHARGING FLOW
TAG NO.: N/A
REF DWG: N/A

Type and Category - N/A

Range - N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

0, 5, 10, 11, 12, 42

REASON:

To Monitor Operation of RCS Injection Systems.

The B&W - designed NSSS does not include a charging system as part of the Emergency Core Cooling System (ECCS). Flow paths from the ECCS to the RCS include high pressure injection (HPI) and low pressure injection (LPI) with the BWST or the RB Sump as the suction source, and the Core Flood Tank injection. HPI and LPI flow rates are monitored, and BWST, RB sump, and Core Flood Tank levels are monitored by RG 1.97 variables. Therefore, Boric Acid Charging Flow does not need to be monitored as a Type D variable to monitor the operation of the ECCS.

This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: HPI FLOW (0-500 gpm)
TAG NO.: MU-23-FI1, MU-23-FI2, MU-23-FI3, MU-23-FI4
REF DWG: D8034039

Type and Category - D, 2

Range - 0 - 500 gpm (Design = 300 gpm)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - N/A - 1 Channel per HPI injection leg.

Power Source - UPS/DG

Display - Indicated in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 8, 9, 15, 17, 37, 42

REASON:

HPI flow measurement is a key variable used to monitor operation of the ESFAS system for design basis events.

PARAMETER:

VARIABLE: HPI FLOW (0 - 200 gpm)
TAG NO.: MU-23-FI5-1, MU-23-FI6-1, MU-23-FI7-1, MU-23-FI8-1, MU-23-FI9, MU-23-FI10, MU-23-FI11, MU-23-FI12
REF DWG: 205-046, MU-11

Type and Category - A, D, 1

Range - 0 - 200 gpm

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes -

Power Source - UPS/DG

Display - Indicated in CR
On Demand in CR, TSC & EOF - Provides input signal to the plant computer system via the Safety Parameter Display System (SPDS)/RECALL

SOURCE:

0, 5, 8, 9, 15, 24, 25, 26, 37, 41, 42, 59, 60

REASON:

HPI flow measurement is a key variable used to monitor operation of the ESFAS system in event of a LOCA.

HPI flow low range (0-200) is a Type A variable because the operator uses this variable to throttle HPI flow in the event of a LOCA. Long term cooling requirements direct the operator to throttle HPI flow to $\leq$545 gpm 72 hours into the accident to ensure long term pump integrity.

PARAMETER:

VARIABLE: FLOW IN LPI SYSTEM
TAG NO: DH-1-FI1, DH-1-FI2
REF DWG: 205-021, DH-01, DH-02

Type and Category - A, D, 1

Range - 0 - 5,000 gpm (Design = 3000 gpm)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - 2 Channels

Power Source - UPS/DG with 1E Standby

Display - Indicated in CR
On Demand in CR, TSC & EOF

SOURCE:

0, 5, 36, 39, 40, 41, 42, 58

REASON:

1. LPI flow measurement monitors LPI flow in post - SBLOCA cooldown, and is used to determine that once LPI flow is established, HPI flow can be terminated. It is also used to throttle LPI flow when operating in piggy-back configuration to assure sufficient NPSH is available for the LPI pump.
2. DH flow measurement monitors LPI safety injection in event of a LOCA, or residual heat removal (RHR) during reactor shutdown.

LPI flow is a Type A variable because it is used by the operator to throttle LPI flow when the ECCS systems are in "piggy back" in order to assure there is sufficient NPSH for the LPI pump.

PARAMETER:

VARIABLE: BORATED WATER STORAGE TANK LEVEL
TAG NO.: DH-7-LI, DH-37-LI, DH-7-LIR-1
REF DWG: 205-021, DH-06, DH-07

Type and Category - A, D, 1

Range - 0 - 50 ft.

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E/DG

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 10, 11, 32, 42

REASON:

BWST level indication is a key variable in that this tank is the primary source of injection water for at least 20 minutes following a LOCA.

The variable is indicated on redundant, qualified, indicators, located on a seismically qualified panel board and one of the redundant channels is recorded.

Due to a lack of seismically qualified panel space the recorder is mounted on panel not seismically qualified.

The recorder itself is non-safety, but environmentally qualified as it is located in a mild environment. It is electrically isolated from the rest of the qualified instrument loop.

B&W's Criteria for BWST sets three criteria which must be met by the BWST. The first criterion is related to fuel handling and transfer operations and is not applicable for accident events. The second criterion requires that sufficient volume be contained in the BWST to provide sufficient time for injection operation prior to switchover to an alternate source. This is a criterion which must be satisfied during normal plant operation to ensure availability of the BWST during an accident. This volume is less than that required to meet the first criterion.

The third criterion is the important one for use during and after an accident. This criterion requires that the BWST level be such that adequate NPSH for all ECCS pumps be available.

To meet the desired intent of the regulatory guide that accident monitoring instrumentation also be used, to the extent practicable, during normal operations, the existing BWST level instrumentation has sufficiently wide range to monitor the level in the BWST. At CR3, the tank level is monitored from 0 to 50 feet. Low and Low-Low alarms are provided and switchover is required to be completed by 7 feet, indicated on DH-7-LI and/or DH-37-LI. Thus, the operator is provided with adequate level indication at all times. The Range is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

BWST Level is a Type A variable because it is used by the operator to initiate swapover of ECCS suction from the BWST to the RB sump during LOCA scenarios.

PARAMETER:

VARIABLE: REACTOR COOLANT PUMP STATUS
TAG NO.: RECALL PT NO. 129, 130, 131, 132, MCB B/M# ICS-ED1, EE1, E1, EF1, EG1, ED10, EE10, EF10, EG10, ED11, EE11, EF11, EG11
REF DWG: N/A

Type and Category - D, 3

Range - 0 - 150% LOAD

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A - 1 per pump

Power Source - OP

Display - Indicating lights in CR
0-150% motor load Indicated in CR
Indicated Total Amps in Switchgear
On Demand in TSC & EOF

SOURCE:

0, 5, 42

REASON:

RC pump motor amps and indicating lights are required to monitor operation of the primary coolant system pumps.

PARAMETER:

VARIABLE: RC SYSTEM SAFETY RELIEF VALVE FLOW/POSITION
TAG NO.: RC-160-MI1, RC-160-MI2, RC-160-MI3
REF DWG: 205-047, RC-17

Type and Category - D, 2

Range - Acoustic System

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A (Sensor Back-up Only)

Power Source - UPS/DG

Display - Indicated in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 42

REASON:

RC System safety valve flow is a key variable to monitor valve operation and loss of primary coolant.

Accelerometers are seismically mounted but are not safety related. This is one of the exceptions to the general design approach for Category 2 instruments.

PARAMETER:

VARIABLE: PRESSURIZER LEVEL
TAG NO.: RC-1-LIR-1, RC-1-LIR-3
REF DWG: 205-047, RC-01, RC-05, RC-06

Type and Category - D, 1

Range - 0 - 320 Inches.

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E/DG

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 1, 2, 5, 10, 11, 12, 42, 59, 60, 61

REASON:

Pressurizer level is a key variable required to ensure proper operation of the pressurizer.

The pressurizer level was sized based on the following. The water volume is chosen such that the reactor coolant system can experience a reactor trip from full power without uncovering the level sensors in the lower shell and to maintain system pressure above the HPI system actuation setpoint. The steam volume is chosen such that the reactor coolant system can experience a turbine trip without covering the level sensors in the upper shell. The range of 0-320" H_2O was based on this criteria and setpoints for automatic or manual actions are based on this range.

The pressurizer is approximately 512 inches tall. The 0 inch reference for the pressurizer level instrument range is 43 inches above the lower datum line (approx. 96 inches from the bottom), 16 inches below the upper set of heaters, and approximately at the level of the second set of heaters. The upper pressurizer level top 320 inches above the 0-inch reference) is 43 inches below the upper datum (approx. 92 inches from the top), and approximately 37 inches from the spray head.

The Range is an NRC accepted exception, per NRC SER Docket No. 50-305, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

Per Deficiency Report (DR) 98-0044, the following component would suffer consequential failure as a result of a pressurizer surge line rupture:

- Instrument tubing for pressurizer level indication instrument RC-1-LT3

Justification for the acceptability of this failure has been provided in SA/USQD No. 99-0255 and included in the Topical Design Basis Document for High Energy Line Breaks Inside Containment (Reference 61).

The Accident Analysis chapters of several B&W Owners Group Utility Final Safety Analysis Reports (SAR), as well as Part II, Volume 2 of the B&W Owners Group Abnormal Transient Operating Guidelines (ATOG) were reviewed to obtain pressurizer level responses to anticipated transients and accidents.

For anticipated transients such as decreasing feedwater temperature, excessive main feedwater flow, loss of main feedwater flow, decreasing steam flow, small steam leaks, loss of external load, loss of off-site power, loss of condenser vacuum and small steam generator tube leaks, the existing ranges for the pressurizer level are sufficient such that indicated level should remain on-scale.

For severe transients (accidents) such as steam line break, steam generator tube rupture and many small break LOCA's, the pressurizer will void. Following ESFAS actuation of the HPI system, actions can be taken as necessary to stabilize the plant. Those actions are based on subcooling margin and RCS pressure, not pressurizer level. For the case of a total loss of feedwater, the pressurizer will go solid unless either main or emergency feedwater is restored to the steam generators within about 15 minutes. Actions taken are dependent on when feedwater is restored, subcooling margin and RCS pressure, not pressurizer level.

In general, for severe transients or accidents, the pressurizer will either void or go solid. A voided pressurizer will cause indicated level to go off-scale low followed by a rapid decrease in RCS pressure to saturation. A solid pressurizer will cause indicated level to go off-scale high accompanied by high RCS pressure, possible large and rapid changes in RCS pressure, PORV and pressurizer safety valve actuation. All of these indications are available in the Control Room.

Based on this information, the existing ranges of pressurizer level indication are sufficient for anticipated transients. For severe transients or accidents, indicated pressurizer level will go off-scale high or low due to the pressurizer going solid or voiding and, as a result, top to bottom instruments would provide no significant additional information. In these cases, subcooling margin, RCS pressure, PORV status and pressurizer safety valve status are monitored to determine actions to be taken.

PARAMETER:

VARIABLE: PRESSURIZER HEATER STATUS
TAG NO.: RC-203-JI, RC-204-JI
REF DWG: 210-654

Type and Category - D, 2

Range - 0 - 1000 kw

Environmental Qualification - Not required

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A

Power Source - Unit Bus with 1E Standby

Display - Wattmeters in CR

SOURCE:

0, 1, 2, 4, 5, 10, 11, 14, 42

REASON:

Pressurizer heater status is important to determine operating status of the pressurizer. Emergency heaters are loaded manually onto the diesels with observation of load before and after.

480V Reactor Auxiliary Bus 3A (MTSW-3C) and 3B (MTSW-3D) are located in the 480 V Switchgear Room in the Turbine Building Elevation 95'-0". No Environmental Zone has been assigned to this area in the E/SQPM Table 4-1 and the Environmental Zone Maps. It must, therefore, be assumed to be a harsh environment for this analysis. Any DBE in this area (Steam Line Break) would also negate the operability of the 480V metal clad switchgear rated for operation per ANSI/IEEE C37.20 at 40°C maximum temperature and atmospheric pressure at a maximum altitude of 6,600 feet. Thus, the requirements for status indication is not valid for DBE conditions in the area since the 480V switchgear is not qualified to function for the DBE condition. This is consistent with the Regulatory Guide 1.97 classification of Type D, Category 2 for the pressurizer heater status indication. Therefore, 10CFR50.49 does not apply.

PARAMETER:

VARIABLE: RC DRAIN TANK LEVEL
TAG NO.: WD-23-LI1
REF DWG: 205-060, WD-04

Type and Category - D, 3

Range - 6" from Bottom - Top

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 42

REASON:

RC drain tank level is required to monitor operation of the RCS system relief valves.

PARAMETER:

VARIABLE: RC DRAIN TANK TEMPERATURE
TAG NO.: WD-24-TI-1
REF DWG: 205-060, WD-02

Type and Category - D, 3

Range - 0 - 400°F

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 10, 11, 42

REASON:

RC drain tank temperature is required to monitor operation of the RCS system relief valves.

Rupture disc (set @ 110 psig) precludes temperature from exceeding 345°F.

RG 1.97 range of 0 - 400°F is acceptable to 50 - 750°F NRC requirement.

PARAMETER:

VARIABLE: RC DRAIN TANK PRESSURE
TAG NO.: WD-22-PI1
REF DWG: 205-060, WD-06

Type and Category - D, 3

Range - 0 - 100 psig (Design = 100 psig)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 42

REASON:

RC drain tank pressure is required to monitor operation of the RCS system relief valves.

PARAMETER:

VARIABLE: STEAM GENERATOR LEVEL
TAG NO.: SP-17-LI1, SP-17-LIR, SP-18-LI1, SP-21-LI1, SP-21-LIR, SP-22-LI1, SP-25-LI1, SP-25-LIR, SP-26-LI1, SP-29-LI1, SP-29-LIR, SP-30-LI1
REF DWG: 205-074, SP-01 thru SP-04

Type and Category - A, D, 1

Range - 0 - 150 Inches (Low Range)
0-100%* (High Range) *Corresponds to 102-394 inches

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E/DG

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 19, 42

REASON:

Steam generator level is a key variable to monitor secondary plant operation (FW, EF, EFIC).

CR3, having a B&W NSSS, utilizes Once Through Steam Generators (OTSG) which produce superheated steam and therefore are not equipped with moisture separators in the steam generator. CR3 installed the Emergency Feedwater Initiation & Control (EFIC) system, which was completed in Refuel 5. This system provides Class 1E, redundant, level indication in the CR. The lower range (start-up) measures 0 to 150 inches and the high range (operating) measures 102 to 394 inches, indicated as 0-100%.

The lower level sensing tap (0 inches) is approximately 6 inches above the lower tube sheet and the upper level sensing tap (394 inches) is at approximately the level of the aspirating ports.

Steam Generator Level is a Type A variable because the operator uses the level to restore OTSG cooling using either Dry OTSG Recovery or Alternate OTSG FW supply.

PARAMETER:

VARIABLE: STEAM GENERATOR PRESSURE
TAG NO.: MS-106-PIR, MS-107-PIR, MS-110-PIR, MS-111-PIR, MS-106-PI1, MS-107-PI1, MS-110-PI1, MS-111-PI1

REF DWG: 205-039, MS-01 and MS-02

Type and Category - A, D, 1

Range - 0 - 1200 psig (Ind)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 4 Channels

Power Source - 1E/DG

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 10, 11, 19, 42

REASON:

Steam generator pressure is a key variable to monitor secondary plant operation.

The steam generator pressure range of 0-1200 psig is acceptable because the safety valve setpoints range from a low of 1050 psig ± 10 psig to 1100 psig ± 10 psig, which are close to 20% above the low setpoint recommendation. The high safety valve setpoint is about 100 psig below the high end of the instrument scale.

The highest safety valve setting is typically 1100 psig. The steam relief capacity is 20-25% above the expected steam flow rate. Excess relief capacity is maintained when safety valves are inoperable. The FSAR analysis indicates a maximum steam pressure of about 1100 psig for operating plants. Based on these facts, it is FPCs position that the existing range of 0-1200 psig is sufficient.

Steam Generator Pressure is a Type A variable because the operator uses it to bypass EFIC isolation actuations when both Steam Generators are <725 psig and they are in the Steam Generator tube Rupture procedure.

PARAMETER:

VARIABLE: MAIN STEAM SAFETY RELIEF VALVE POSITION
TAG NO.: MSX-1, MSX-2, MSX-3, MSX-5
REF DWG: 209-039, MS-23

Type and Category - D, 2

Range - Video Display

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A

Power Source - UPS/DG

Display - Video display unit viewing MSSV vent stacks is mounted on the MCB

SOURCE:

0, 1, 2, 4, 5, 10, 11, 27, 42, 53

REASON:

Main steam relief valve position is important to monitor secondary plant releases.

The video cameras show all the Main Steam Safety Valves and the Atmospheric Dump Valves.

The video cameras and the closed circuit screen are non-safety. This is one of the exceptions to the general design approach for category 2 instruments.

PARAMETER:

VARIABLE: MAIN FEEDWATER FLOW
TAG NO.: SP-8A-FI1, SP-8A-FI2, SP-8B-FI2, SP-8A-FIR1, SP-8B-FI1
REF DWG: D8034031, Sht. 1 and 3

Type and Category - D, 3

Range - 0 - 6,000,000 lb/hr (Design = 5.5×10^6 lbs/hr)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A - 3 Channels

Power Source - UPS/DG

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 42

REASON:

Main feedwater flow is important to monitor secondary plant operation during normal operation.

PARAMETER:

VARIABLE: EMERGENCY FEEDWATER FLOW
TAG NO.: EF-23-FI1, EF-24-FI1, EF-25-FI1, EF-26-FI1
REF DWG: 205-026, EF-01 and EF-02

Type and Category - D, 1

Range - 0 - 1000 gpm (Design = 740 gpm)

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 4 Channels

Power Source - 1E

Display - Indicated and Recorded on demand in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 42

REASON:

Emergency feedwater flow is important to monitor secondary plant operation during a transient.

A redundant 4-channel system with all safety parts seismically qualified, and transmitters environmentally qualified were installed in conjunction with the EFIC modifications.

PARAMETER:

VARIABLE: EFP-2 FLOW (0-700 GPM)
TAG NO.: EF-62-FI
REF DWG: 205-026, EF-10

Type and Category - D, 2

Range - 0 - 700 gpm (Design = 550) Note 1

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy -N/A

Power Source - UPS/DG

Display - Indicated in the CR

SOURCE:

38

REASON:

Note 1:
This indication is in addition to the EFW flow indication on each leg to the steam generators. This is an indication of EFP-2 operation only, and as such, does not meet the following criteria of TABLE 3 of the Regulatory Guide: 1) It is not a category 1 because it is not the primary redundant indication noted for "B&W plants". 2) Range is more than 110% of minimum required flow, plus instrument error, which could be considered "maximum flow anticipated in normal operation" (RG note 11).

PARAMETER:

VARIABLE: EMERGENCY FEEDWATER TANK LEVEL
TAG NO.: EF-98-LI1, EF-99-LI1
REF DWG: 205-026, EF-05

Type and Category - A, D, 1

Range - 0 - 38 FT.

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E

Display - Indicated and Recorded on Demand in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 28, 42

REASON:

Emergency feedwater tank level is a key variable to ensure water supply for emergency feedwater.

Emergency feedwater tank level is a Type A variable because it is used by the operator to initiate the change in Emergency feedwater suction source to either the condensate storage tank or the hotwell. (alternate EFW sources).

PARAMETER:

VARIABLE: CONTAINMENT SPRAY FLOW
TAG NO.: BS-1-FI1, BS-1-FI2
REF DWG: D8034036

Type and Category - D, 2

Range - 0 - 1800 gpm (Design = 1500 gpm)

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A - 2 Channels

Power Source - UPS/DG

Display - Indicated in CR

SOURCE:

0, 5, 42, 57

REASON:

Containment spray flow is important to monitor operation of the Reactor Building spray system in event of an accident.

PARAMETER:

VARIABLE: RB FAN HEAT REMOVAL
TAG NO.: Computer Pt. S348, S376, S387 (SW-47-FT1, SW-51-FT1, SW-55-FT1), ESF-A-AW, ESF-A-BJ, ESF-B-AW, ESF-B-BJ (ES STATUS LIGHTS)
REF DWG: 205-056, SW-01; 208-028, ES-A27 and ES-B27

Type and Category - D, 2

Range - On-Off Indicator Lights
NSCCW Flow

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A

Power Source - VBDP-2

Display - ES Status Lights (Display for SW flow to RBCU is available on demand via plant computer.)

SOURCE:

0, 5, 13, 42

REASON:

RB heat removal is important to monitor Reactor Building cooling in event of an accident.

The following position is a justification developed by the BWOG Reg. Guide 1.97 Task Force.

The plant has a design air flow rate from the Reactor Building fans during normal and accident or emergency conditions. The design flow rates are achieved by reducing the normal running speed of the fan motors by about one-half during accidents where the heavier steam-air mixture might overload the motors at full speed. The fan cooling units are cooled by cooling water from the Nuclear Services Closed Cycle Cooling System (SW).

For the following reasons, the status of the fan breakers and cooling water flow rates are the measured variables. The primary indication that the Reactor Building is being cooled is the Reactor Building temperature. A first indication that the Reactor Building fans are performing their function is an indication of the status of the fan breakers to ensure that the fans are on and the delivery of cooling water flow to the cooling units. The flow variable was upgraded to comply with RG 1.97 requirements during Refuel 6.

As backup information to ensure coupling between the fan and motor, each fan is equipped with vibration detectors which annunciate in the Control Room. Calibrated percent load meters for the motors are also located in the Control Room.

PARAMETER:

VARIABLE: CONTAINMENT ATMOSPHERE TEMPERATURE
TAG NO.: AH-536-TIR, AH-537-TIR, AH-538-TIR, AH-539-TIR
REF DWG: 205-005, AH-01

Type and Category - D, 2

Range - 0 - 400°F

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A - 4 Measurements

Power Source - UPS/DG

Display - Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 19, 42

REASON:

Containment atmospheric temperature is important to indicate accomplishment of cooling following an accident.

The temperature elements are non-safety. This is one of the exceptions to the general design approach for category 2 instruments.

PARAMETER:

VARIABLE: CONTAINMENT SUMP WATER TEMPERATURE
TAG NO.: N/A
REF DWG: N/A

Type and Category - N/A

Range - N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

0, 5, 10, 11, 42

REASON:

The NRC RG 1.97 requires containment sump water temperature indication as a Type D variable for the purpose of monitoring the operation of containment cooling systems. No additional justification is provided.

It is expected that this information would be used following high energy line breaks in containment. While containment sump temperature trends may be indicative of high energy fluid leakages and containment cooling, it would be difficult to conceive of any correlation from monitored values to any useful measure of success.

Containment sump temperatures impact containment cooling only when the Reactor Building spray system is in operation with suction being taken from the sump. This would be expected to be used only after depletion of available supplies from the BWST.

a. Containment Cooling System Monitoring

Containment atmospheric temperature instrumentation provides the most direct indication of containment cooling system success. Containment atmospheric temperature instrumentation was upgraded during Refuel 6 to meet RG 1.97 requirements.

The next most valuable indication of containment cooling is provided by instrumentation which monitors the operation of systems with a containment cooling function. This function is provided by the Reactor Building Spray System (BS) and the Reactor Building Air Handling System (AH).

The Reactor Building containment Fan Heat Removal Cooling Water Flow Instrumentation (SW) was upgraded to provide heat removal indication meeting the requirements of RG 1.97 during Refuel 6 (See position Page 50).

Containment atmospheric temperature is recorded in the Control Room. The Reactor Building air handling fan motor breaker positions, indicating lights and percent full load ammeter indicators representative of air flow loading are monitored on the control board. Fan cooling water flow leakage is also monitored and alarmed.

Containment sump water temperature provides only a crude indication of containment cooling system success. Because of this and the availability of the instrumentation described above, sump water temperature instrumentation is not necessary for containment cooling system monitoring. Nevertheless, containment sump temperature can be determined when the LPI is in the recirculation mode, using temperature indicators meeting all other RG 1.97 requirements.

b. Equipment Temperature Limits

Protection of DH and BS from Excessive Sump Temperatures: These systems are designed for fluid temperatures in excess of the RG 1.97 required range for sump water temperature instrumentation (Ref: FSAR, Table 6-3). No operator action is required in response to sump water temperature. Actual options available with excessive sump water temperatures would be limited to the reactor coolant system and containment cooldown prior to transferring to the recirculation mode of containment spray. This transfer is not required for over an hour after a LOCA, in which time the sump temperature is below 205°F.

c. NPSH Requirements

The minimum available NPSH for the Decay Heat Removal pumps is conservatively calculated with sufficient safety margin such that indication of sump temperature is not required in order to insure adequate NPSH and no automatic or manual actions are initiated based on this temperature.

PARAMETER:

VARIABLE: MAKEUP FLOW-IN
TAG NO.: MU-24-FI
REF DWG: 205-041, MU-06

Type and Category - D, 3

Range - 0 - 200 gpm (Design = 115 gpm)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated in CR

SOURCE:

0, 5, 10, 11, 42

REASON:

The following position is a justification developed by the BWOG Reg. Guide 1.97 Task Force.

During design basis events such as LOCAs, the Makeup and Purification System (MU) is isolated. Makeup flow is a backup variable to the makeup line isolation valve position. During normal operation and certain design basis events, the MU System is used to supply borated makeup water into the RCS to balance letdown flow out of the RCS. It also adds makeup water in order to maintain pressurizer level at its setpoint. Thus, makeup flow is an important variable for monitoring the operation of the MU System. For the reasons provided in the Position Section for the variable, Makeup Tank Level (Page 57), it is suggested that this variable can be a backup to Makeup Tank Level. As a backup Type D variable, it is appropriate that Makeup Flow be classified Category 3. This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: LETDOWN FLOW-OUT
TAG NO.: MU-4-FI
REF DWG: 205-041, MU-05

Type and Category - D, 3

Range - 0 - 160 gpm (Design = 140 gpm)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 10, 11, 42

REASON:

The following position is a justification developed by the BWOG Reg. Guide 1.97 Task Force.

During design basis events such as LOCAs, the MU System is isolated. Letdown flow is a backup variable to the letdown isolation valve position. During normal operation and certain design basis events such as small break LOCAs, the MU System is used to supply borated makeup water into the RCS to balance letdown flow out of the RCS. Thus, letdown flow is an important variable for monitoring the operation of the MU System. For the reasons provided in the position section for the variable Makeup Tank Level (Page 57), it is suggested that this variable can be a backup to Makeup Tank Level. As a backup Type D variable, it is appropriate that letdown flow be classified Category 3. This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

For CR3, normal letdown flow rate through the block orifice is 45 gpm with a maximum flow rate of 140 gpm with both letdown coolers in operations. Having this maximum flow rate of 140 gpm the range of letdown flow indicator is 0 to 160 gpm which adequately meets the Regulatory Guide recommendation of 0 to 110% design flow.

PARAMETER:

VARIABLE: MAKEUP TANK LEVEL
TAG NO.: MU-14-LIR1
REF DWG: 205-041, MU-07;
D8034039, Sht. 2

Type and Category - D, 2

Range - 0 - 120 Inches

Environmental Qualification - Yes - SR portion

Seismic Qualification - Yes - SR portion

Quality Assurance - Yes - SR portion

Redundancy - N/A - 2 Channels

Power Source - UPS

Display - Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 42

REASON:

The following position is a justification developed by the BWOG Reg. Guide 1.97 Task Force.

During normal operation and certain design basis accidents where the MU System is still operable, the Makeup Tank Level is the key variable used to provide indication that the MU System is operating properly. Makeup Tank Level information provides the first indication that a suction source for the Makeup pumps is available. Since the Makeup Tank is a surge volume for the RCS, Makeup Tank Level and Pressurizer Level indications can be used to qualitatively assess Makeup Flow into the RCS and Letdown Flow from the RCS.

Quantitative indication of Makeup Flow and Letdown Flow can be provided by flow instrumentation for these variables. However, in most instances, it is more important to know that Makeup and/or Letdown is established (qualitative) and not necessarily what those flow rates are (quantitative) in order to determine the operation of the MU System. Since Pressurizer Level instrumentation is Category 1 and the suggested Makeup Tank Level instrumentation be Category 2, then high quality instrumentation is available to provide information on the status and operation of the MU System. Flow rate indication provided for Makeup Flow and Letdown Flow can be used as confirmatory, backup information to Makeup Tank Level and Pressurizer Level.

Meets intent of RG 1.97, 2 ½" from bottom to 4" from top of vessel. Parts of safety system are seismic with QA. QA requirements meeting CR3 licensing commitments were applied to safety related portions of this instrument string.

PARAMETER:

VARIABLE: COMPONENT COOLING WATER TEMPERATURE TO ESF SYSTEMS
TAG NO.: DC-35-TI, DC-39-TI, RW-12-TI, RW-13-TI, RW-19-TI, RW-32-TI, RW-33-TI, RW-43-TI, SW-132-TI
REF DWG: 208-019, DC-02; 205-050, RW-01

Type and Category - D, 2

Range - 0 - 200°F (DC Sys)
0 - 250°F (SW Sys)

Environmental Qualification - N/A - Mild Environment

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated in CR

SOURCE:

0, 5, 42

REASON:

Component cooling temperatures are important to monitor operation of ESF cooling systems.

The 0-200°F range is for the Decay Heat Closed Cycle Cooling Systems (DC) and the 0-250°F range is for the Nuclear Services Closed Cycle Cooling systems (SW).

This equipment was originally purchased without Quality Assurance documentation. Future equipment will be purchased with the requirement to specify the applicable Quality Assurance practices. The temperature elements are non safety. This is one of the exceptions to the general design approach for Category 2 instruments.

The Component Cooling Water to ESF system at CR3 is provided by several systems, Nuclear Services Closed Cycle Cooling System (SW), Decay Heat Closed Cycle Cooling System (DC), and Nuclear Service and Decay Heat Sea Water System (RW). The SW and DC systems are closed cycle systems that provide the direct component cooling. The RW system provides cooling to the SW and DC systems and is an open system rejecting heat to the ultimate heat sink, the Gulf of Mexico.

SW Temperature - SW temperature is taken from the common discharge header of the SW Heat Exchangers, SWHE-1A, B, C, and D. The indicator, SW-132-TI, is a Bailey RY meter, 0-250° F, located on the ESF section of the Main Control Board.

DC Temperature - DC-35-TI and DC-39-TI are Bailey RY meters, 0-200° F, located on the ESF panels on the Main Control Board. These indicators provide DC closed cycle coolers outlet temperature.

RW Temperature - There are 6 RW temperature indicators on the ES section of the Main Control Board. RW-12-TI and RW-13-TI are one half of dual Bailey RY meters, with a 50-150°F range, and indicate the inlet side of the Decay Heat Closed Cycle Heat Exchangers, the RW side. The other half of the Bailey meters are for RW-32, 33-TE, which are the outlet temperatures. With the inlet and outlet temperature located next to each other, the temperature difference across the heat exchangers is easily seen. A similar setup on the SW heat exchangers temperature exists with RW-19, 43-TE.

PARAMETER:

VARIABLE: COMPONENT COOLING WATER FLOW TO ESF SYSTEMS (SYSTEM STATUS)
TAG NO.: DC-5-PI, DC-6-PI, DC-50-LI, DC-54-LI, SW-2-PI, SW-139-LI
REF DWG: 205-019, DC-01; 205-056, SW-02

Type and Category - D, 2

Range - 0-12 feet (normal operating band is 8.5 to 10 feet indicated) (SW Sys);
0-14 feet (normal operating band is 8.5 to 11.2 feet indicated) (DC Sys);
0 - 200 psig (SW Sys);
0 - 60 psig (DC Sys)

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A

Power Source - VBDP-3 (UPS/DG)
VBDP-4

Display - Indicated in CR (SW temperature display is available on demand via plant computer)

SOURCE:

0, 1, 5, 10, 11, 13, 42

REASON:

Component cooling systems status is important to monitor operation of the ESF cooling systems.

There are presently no flow indications on the main control board for Decay Heat Closed Cycle Cooling (DC) and Nuclear Services Closed cycle Cooling (SW) systems. Local flow indication for these systems is available. Indicated flow measurements in the Control Room are not deemed necessary because the DC and SW Systems surge tank levels provide better information to the operator. The wide range of design flows to various ESF components would not necessarily be representative of overall system performance. Service water header pressures and remote actuated valve positions are available to the operator and along with the surge tanks levels, which provide a better overall indication of system status. This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

QA requirements meeting CR3 licensing commitments were applied to safety related portions of this instrument string.

PARAMETER:

VARIABLE: HIGH-LEVEL RADIOACTIVE LIQUID TANK LEVEL
TAG NO.: WD-49-LI, WD-54-LI, WD-58-LI, WD-62-LI
REF DWG: 308-818

Type and Category - D, 3

Range - 0 - 100%

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - Instrument Air

Display - Local WD Panel (As described in Reason below.)

SOURCE:

0, 5, 42, 43

REASON:

Level indication of concentrated waste tanks, concentrated boric acid tanks, and spent resin tank is important to indicate storage volume.

Tanks covered by this variable are:

RCS Bleed Tank (3)
Misc. Waste Storage Tank (1)

The level indication for the Misc. Waste Storage Tank is indicated on the radioactive waste disposal control panel located in the Auxiliary Building. High level alarms at this panel will cause a common alarm to actuate on the main control board. The controls for the liquid waste disposal system are all located at the local panel; therefore, indication on the main control board would not enhance operator control from the Control Room.

The level indication for the RC Bleed Tanks are indicated on the Main Control Board.

PARAMETER:

VARIABLE: RADIOACTIVE GAS HOLD-UP TANK PRESSURE
TAG NO.: WD-16-PI, WD-17-PI, WD-18-PI
REF DWG: 308-806

Type and Category - D, 3

Range - 0 - 150 psig (Design = 150 psig)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - Instrument Air

Display - Local WD Panel

SOURCE:

0, 5, 10, 11, 42

REASON:

Waste gas holdup tank pressure is important to indicate storage capacity.

The control and indications for the waste disposal system are located on the radioactive waste disposal panel in the Auxiliary Building. Indication of radioactive gas hold-up tank pressure is not a necessary Control Room variable for the post accident monitoring. In the event of an accident which results in significant failed fuel or significant radioactive gas release, the manual transfer of radioactive gases to the radioactive gas hold-up tanks would not be attempted since the Reactor Building would be utilized as the hold-up tank. There are no automatic transfer operations involving the radioactive gas hold-up tanks during post-accident conditions is not necessary since these tanks are not utilized for accident mitigation.

The radioactive gas hold-up tanks are equipped with relief valves which are set at 125 psig. The range of the pressure indication is 120% above the relief valve setting. This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: EMERGENCY VENTILATION DAMPER POSITION INDICATION
TAG NO.: DAMPER OPEN AND CLOSED INDICATION FOR AHD-17, AHD-22, AHD-12, AHD-12D, AHD-2C, AHD-2E, AHD-1C, AHD-1E, & AHD-3.
REF DWG: 308-847

Type and Category - D, 2

Range - On - Off Fan Lights; OP-CL Lights

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated Lights in CR

SOURCE:

0, 5, 10, 11, 13, 29, 42, 64

REASON:

Dampers covered under this category are those used in ventilation systems for the following:

Emergency Diesel Generator Rooms
Auxiliary Building
Control Complex
Decay Heat Pump Area
Spent Fuel Cooling Pump Area

The Control Complex Dampers status are indicated off the Limit Switches located on their respective damper (see table below). These red and green lights are located on the rear of the Main Control Board in the demarcation area titled Control Complex "A". The limit switches are non-safety. This is an NRC accepted exception.

Damper Tag Number	Closed End Limit Switch	Open End Limit Switch
AHD-17	AH-747-ZS	NONE
AHD-22	AH-748-ZS	NONE
AHD-12	AH-746-ZS1A	AH-746-ZS1B
AHD-12D	AH-1029-KS1A	AH-1029-KS1B
AHD-2C	AH-382-KS1A	AH-382-KS1B
AHD-2E	AH-1030-KS1A	AH-1030-KS1B
AHD-1C	AH-384-KS1A	AH-384-KS1B

AHD-1E	AH-910-KS1A	AH-910-KS1B
AHD-3	AH-383-KS1A	NONE

The dampers in all these systems are controlled from the fan start circuitry and do not have individual control switches. The Control Complex Habitability Boundary Dampers are positioned to their isolation position also by an ES, RM-A5, or Toxic Gas Monitor Signal. Only the Control Complex Dampers have individual damper position indication. Panel lights show when the fan circuitry is operating.

Back-up operational data is provided to operators by high quality commercial grade low flow and high temperature alarms. The control complex dampers also have open position lights.

The above data should be adequate to determine if an HV system is operational. Individual damper position would only be beneficial if isolation were required.

PARAMETER:

VARIABLE: STATUS OF STANDBY POWER

TAG NO.:
1. AC BUSES
   - A. 4160V ES Bus Voltmeter SSF-DT (3A) SSF-ED (3B)
   - B. 480V ES Bus Voltmeter SSF-FB (3A) SSF-FN (3B)
2. 250/125 VDC Bus 3A & 3B Indication lights
   COF-AF (A+) COF-AJ (A-) COF-AC (B+) COF-AI (B-)
3. Inverters 3A, B, C, D
   COF-AH (A) COF-AE (B) COF-AG (C) COF-AD (D)
4. EDG 3A & 3B
   Freq. SSF-GZ (A) SSF-HA (B)
   Var SSF-AG (A) SSF-AW (B)
   *Watt SSF-AH (A) SSF-AX (B)
   KV SSF-AJ (A) SSF-AY (B)
   Amp SSF-AK (A) SSF-AZ (B)

NOTE: *Wattmeters SSF-AH(A) & SSF-AX (B) are A, 1 variables

REF DWG: N/A

Type and Category - D, 2; A, 1

Range - CR3 -

DG	INVERTER	4160V	480V	250/125VDC
3A, 3B	3A to 3D	3A, 3B	3A, 3B	3A, 3B Power
Volts Amps	Pwr Available	Volts	Volts	Available
*Kilowatts	Ind. Lts - White			Ind. Lts -White

NRC - Plant Specific

Environmental Qualification - Yes	-	Yes	No	-
Seismic Qualification - No not required.				
Quality Assurance - No	-	No	No	-
Redundancy - Redundancy Based on Dual Buses				
Power Source - UPS	-	UPS	UPS	-
Display - Indicated in CR				

SOURCE:

0, 5 , 42, 44

REASON:

Electrical meters for DGs, inverters and vital buses are important to monitor electrical system status. *EGDG Wattmeters in Main Control Board are A, 1 variable because they give direct input to operators for control of diesel generator loading.

PARAMETER:

VARIABLE: CONTAINMENT AREA RADIATION - HIGH RANGE
TAG NO.: RM-G29-RIR, RM-G30-RI, RM-G29-RI
REF DWG: 205-049, RM-10 and RM-10A

Type and Category - C, E, 1

Range - 1 to 10^8 R/hr

Environmental Qualification - Yes

Seismic Qualification - Yes

Quality Assurance - Yes

Redundancy - Yes - 2 Channels

Power Source - 1E

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 4, 5, 42, 48, 49, 50, 51

REASON:

Containment high range radiation monitors are important to detect and assess significant releases, and for emergency planning.

I & C calculation I89-0006, Rev. 4 demonstrated that the response accuracy of Radiation Monitors RM-G29 and RM-G30 on the lowest scale (10^0) could result in a condition where the actual radiation level is greater than 100% above the indicated value. The 10^0 scale is for radiation dose between 1 Rad/hr and 10 Rad/hr. The greater than 100% response inaccuracy is outside the "factor of 2" range (-50% to +100%) recommended by Regulatory Guide 1.97. The 10^1 through 10^8 ranges of Radiation Monitors RM-G29 and RM-G30 are within the response accuracy "factor of 2" range recommended by Regulatory Guide 1.97.

PARAMETER:

VARIABLE: RADIATION EXPOSURE RATE INSIDE BUILDING OR AREAS WHERE ACCESS IS REQUIRED TO SERVICE EQUIPMENT IMPORTANT TO SAFETY

TAG NO.: RM-G4-RIR, RM-G9-RIR, RM-G10-RIR, RM-G1-RIR, RM-G2-RIR, RM-G6-RIR, RM-G3-RIR, RM-G17-RIR

REF DWG: 205-049, RM-02, RM-03, RM-04

Type and Category - E, 3

Range - 0.0001 to 10 R/hr

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - 1E

Display - Indicated and Recorded on MCB

SOURCE:

0, 5, 10, 11, 42

REASON:

The following position is a justification developed by the BWOG Reg. Guide 1.97 Task Force.

NRC RG 1.97, Rev. 3 requires area radiation monitors inside buildings or areas where access is required to service equipment important to safety. The NRC identified purposes for this instrumentation are: "Detection of Significant Releases, Release Assessments, and Long Term Surveillance." This is a Type E variable with the overall purpose of being monitored as required in determining the magnitude of the release of radioactive materials and continually assessing such releases. The required range for these monitors is 0.1 to 10^4 R/hr.

RG 1.97 describes areas of concern as those where access is required to service safety related equipment. This implies that this instrumentation may be used for purposes other than those described above, i.e., for Health Physics Purposes.

For purposes of determining the magnitude of releases, the area radiation exposure rate monitors are clearly of very minor importance. Determination of release magnitude is done by other Type E variables associated with release paths. There is no useable correlation between area exposure rate monitors and amount of release.

Detection of significant releases by area radiation exposure rate monitoring is secondary to that provide by the release path monitoring. Nonetheless, area radiation levels inside the plant are

monitored to verify compliance with 10CFR20. These instruments are considerably more sensitive (1000x) than required by RG 1.97 and are sufficient for supporting the detection of significant releases.

Determinations of accessibility of equipment for service or long term surveillance is the function of health physics personnel, generally using portable instrumentation. Monitoring of recordings of area radiation exposure rates from the Control Room is not a substitute for this health physics function. However, exposure rate monitoring equipment in areas outside containment have an upper range of 10 R/hr, which is adequate for initial assessments of accessibility.

These ranges are based on background reading in the areas in which they are located. Should personnel entry be required in areas where these monitors have gone off scale or indicate a high radiation area a Health Physics Escort would accompany personnel into these areas using portable instrumentation to assess radiation levels. The high range for portable instrumentation at CR3 is 10^3 R/hr. We do not anticipate even under emergency conditions, sending personnel into radiation fields of this magnitude.

This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: CONTAINMENT OR PURGE EFFLUENT, NOBLE GAS
TAG NO.: AH-717-FIR, RM-A1-RIR-2 and RM-AI-RY4 (Low Range), RM-AI-RI4 (Mid Range), RM-AI-RI5 (High Range)
REF DWG: 205-005, AH-04; 205-049, RM-06 and RM-06A

Type and Category - C, E, 2

Range – 0 -65,000 cfm (Design = 50,000 CFM)

Low Range: 2×10^{-6} to 1×10^{-2} micro curies/cc Kr-85
1×10^{1} to 1×10^{6} cpm (Indicated)

Mid Range: 1×10^{-3} to 1×10^{2} micro curies/cc Xe-133
1×10^{-2} to 1×10^{3} mr/hr (Indicated)

High Range: Background to 1×10^{5} micro curies/cc Xe-133
1×10^{-1} to 1×10^{7} mr/hr (Indicated)

Environmental Qualification - No (Mild Environment)

Seismic Qualification - N/A

Quality Assurance - No - (Radiation monitoring equipment was originally purchased without Quality Assurance documentation. Future purchases will specify QA requirements.)

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated and Recorded in CR

SOURCE:

0, 5, 42

REASON:

Noble gas concentration and vent flow rate is required to detect a breach of containment and significant releases.

The sensors and indicators are non-safety. This is one of the exceptions to the general design approach for Category 2 Instruments.

PARAMETER:

VARIABLE: REACTOR SHIELD BUILDING ANNULUS
TAG NO.: N/A
REF DWG: N/A

Type and Category - N/A

Range - N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

0, 5, 42

REASON:

N/A in CR3 design.

PARAMETER:

VARIABLE: AUXILIARY BUILDING NOBLE GAS
TAG NO.: AH-32-FIR, RM-A2-RIR-2 and RM-A2-RY4 (Low Range), RM-A2-RI4 (Mid Range) RM-A2-RI5 (High Range)
REF DWG: 205-005, AH-05; 205-049, RM-07A

Type and Category - C, E, 2

Range - 0-200,000 cfm (Design = 156,680 cfm)
Low Range: 2×10^{-6} to 1×10^{-2} micro curies/cc Kr-85
1×10^{1} to 1×10^{6} cpm (Indicated)
Mid Range: 1×10^{-3} to 1×10^{2} micro curies/cc Xe-133
1×10^{-2} to 1×10^{3} mr/hr (Indicated)
High Range: Background to 1×10^{5} micro curies/cc Xe-133
1×10^{-1} to 1×10^{7} mr/hr (Indicated)

Environmental Qualification - No (Mild Environment)

Seismic Qualification - N/A

Quality Assurance - No - (The radiation monitoring equipment was originally purchased without Quality Assurance documentation. Future radiation monitoring equipment will be purchased with the requirement to specify the applicable Quality Assurance practices.)

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated and Recorded in CR
-Concentration also displayed on demand in the TSC and EOF common plant vent, Category E2, from Auxiliary Building.

SOURCE:

0, 5, 42

REASON:

Noble gas concentration and vent flow rate is required to detect a breach of containment and significant releases.

PARAMETER:

VARIABLE: CONDENSER AIR REMOVAL SYSTEM EXHAUST
TAG NO.: RM-A12-RIR, RM-A12-RY2
REF DWG: 205-049, RM-05

Type and Category - C, E, 2

Range - 2×10^{-6} to 1×10^{-2} micro curies / cc Kr-85
1×10^{-1} to 1×10^{6} cpm (indicated)

Environmental Qualification - No (Mild Environment. See below.)

Seismic Qualification - N/A

Quality Assurance - No - (The radiation monitoring equipment was originally purchased without Quality Assurance documentation. Future radiation monitoring equipment will be purchased with the requirement to specify the applicable Quality Assurance practices.)

Redundancy - N/A

Power Source - 1E

Display - Indicated and Recorded in CR
On Demand in TSC & EOF

SOURCE:

0, 5, 10, 11, 42

REASON:

Noble gas monitor in condenser air removal system exhaust is the key variable for detection of a breach of the primary to secondary loop boundary. The Auxiliary Building RM and flow meters are important to detect significant releases.

The condenser air removal system exhausts through the Auxiliary Building in which the flow is monitored. The range of the monitor in the Auxiliary Building is relied upon to satisfy the range requirement of RG 1.97 for condenser air removal system exhaust.

The range was corrected to confirm to the requirements of NUREG-0737, Item II.F.1.1.

The RM-A12 sensor is located in the Turbine Building, and since that environmental zone is undefined and not a harsh environment except for a Steam Line Break in the Turbine Building (when RM-A12 is not needed), the RM-A12 sensor is considered to be qualified for the environment it will see when required to perform its function. The sensors and indicators are non-safety. This is one of the exceptions to the general design approach for Category 2 instruments.

PARAMETER:

VARIABLE: COMMON PLANT VENT
TAG NO: N/A
REF DWG: N/A

Type and Category - N/A

Range - N/A

Environmental Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

REASON:

See Auxiliary Building, Noble Gas

PARAMETER:

VARIABLE: VENT FROM SG SAFETY VALVES OR ADVs
TAG NO.: RM-G25-RI, RM-G28-RI, RM-7-RR
REF DWG: 205-049, RM-08

Type and Category - E, 2

Range - 8.7×10^{-3} to 2.5×10^{7} m ci /cc Xe133

Environmental Qualification - Yes

Seismic Qualification - N/A

Quality Assurance - Yes

Redundancy - N/A - 1 each ADV

Power Source - 1E

Display - Indicated in CR
Recorded On Demand

SOURCE:

0, 1, 2, 5, 10, 11, 30, 42

REASON:

The four 24" main steam headers contain a total of 16 relief valves and 2 atmospheric dump valves. Each atmospheric dump valve discharge is monitored for radiation by monitors with readouts in the Control Room. The system was calibrated in terms of microcurie/cc Xe133 in order to comply with NUREG-0737. Radioactive releases are manually calculated.

This variable is only used during a S.G. tube rupture type accident. The results of this accident do not create a harsh environment. Therefore, they meet the environmental qualifications for the normal environment.

The indicators and sensors are non-safety. This is one of the exceptions to the general design approach for Category 2 Instruments.

PARAMETER:

VARIABLE: ALL OTHER IDENTIFIED RELEASE POINTS
TAG NO.: N/A
REF DWG: N/A

Type and Category - N/A

Range - N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

REASON:

There are no other identified release points. See Auxiliary Building, Noble Gas and Containment or Purge Effluent, Noble Gas.

PARAMETER:

VARIABLE: ALL PLANT RELEASE POINT - PARTICULATES AND HALOGENS
TAG NO.: AH-717-FIR, AH-32-FIR
REF DWG: N/A

Type and Category - E, 3

Range – RB: Radiation – NA – Sample Flow - 0-65,000 cfm (RB)
(Design = 50,000 cfm)
AB: Radiation – NA – Sample Flow - 0-200,000 cfm (AB)
(Design = 156,680 cfm)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Flow is Indicated and Recorded in CR

SOURCE:

0, 5, 42

REASON:

To provide information regarding release of radioactive halogens and particulates.

Particulate and Halogen Filters from RM-A1 and RM-A2 can be removed and analyzed offsite. This provides a range for radioactivity concentration in excess of that specified in Regulatory Guide 1.97.

PARAMETER:

VARIABLE: AIRBORNE RADIO HALOGENS AND PARTICULATES
TAG NO.: N/A
REF DWG: N/A

Type and Category - E, 3

Range - 10^{-9} to 10^{-3} microcurie/cc

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - Vital Bus - Portable Samplers are powered from either local 120VAC outlets or batteries. The spectrometer is powered from a vital buss; i.e. battery backed.

Display - LAB ONLY

Other - Portable sampling and onsite analysis.

SOURCE:

0, 5, 42

REASON:

To estimate release rates of radioactive materials during an accident.

Various portable air samplers can be used to obtain the sample which is then taken to the Lab for counting. (Such as the Radevco H809 high volume air sampler.)

FPC also has portable particulate monitors and mini-scalers on hand. Specific makes and models of current equipment is maintained in Health Physics Procedure 409, Inventory and Availability of Emergency Supplies/Equipment.

Once the sample is at the Lab, we have multi-channel gamma-ray spectrometer systems to provide the capability of onsite analysis.

PARAMETER:

VARIABLE: PLANT AND ENVIRONS RADIATION
TAG NO.: N/A
REF DWG: N/A

Type and Category - E, 3

Range - 10^{-3} to 10^{3} R/hr (Exception)

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - Batteries

Display - Portable

SOURCE:

0, 5, 10, 11, 42

REASON:

To monitor radiation in plant and environs where range of normal monitor impractical for accident levels.

Personnel not permitted in areas exceeding 10^{3} R/hr.

Range of portable monitors is acceptable deviation to NRC required range of 10^{-3} to 10^{4} R/hr.

PARAMETER:

VARIABLE: PLANT AND ENVIRONS RADIOACTIVITY
TAG NO.: N/A
REF DWG: N/A

Type and Category - E, 3

Range - Multi-channel Gamma-Ray Spectrometer

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - Batteries

Display - 2 Channel portable gamma ray spectrometers are available (Eberline SAM-2). Also a mobile multi-channel analyzer/computer contracted with Dept. of Health and Rehabilitation.

SOURCE:

0, 5, 42

REASON:

To monitor airborne radioactivity in the plant and environs where range of normal monitor is impractical for accident levels.

PARAMETER:

VARIABLE: WIND DIRECTION
TAG NO.: MM-13-MI, MM-18-SR, MM-14-MI, MM-19-SR
REF DWG: 205-070, MM-01 and MM-02

Type and Category - E, 3

Range - 0 - 360°

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated and Recorded in CR

SOURCE:

0, 5, 42

REASON:

To assess impact of atmospheric releases.

PARAMETER:

VARIABLE: WIND SPEED
TAG NO.: MM-11-SI, MM-18-SR, MM-12-SI, MM-19-SR
REF DWG: 205-070, MM-01 and MM-02

Type and Category - E, 3

Range - 0 - 50 M/sec

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated and Recorded in CR

SOURCE:

0, 5, 42

REASON:

To assess impact of atmospheric releases.

PARAMETER:

VARIABLE: ESTIMATION OF ATMOSPHERIC STABILITY
TAG NO.: MM-15-TI, MM-16-TI, MM-17-TI, MM-20-TR, MM-21-TR
REF DWG: 205-070, MM-04

Type and Category - E, 3

Range - -5° to +10°F

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - UPS/DG

Display - Indicated and Recorded in CR

SOURCE:

0, 5, 10, 11, 42

REASON:

To assess impact of atmospheric releases.

In accordance with Regulatory Guide 1.23, Table 1, the measurement of temperature difference for estimating atmospheric stability requires a range from -1.9°C to +4.0°C for the 100 meter height. The height distance between temperature measuring points at CR3 is 142 ft. At this distance the RG 1.23 equivalent range of required temperature to estimate stability in degrees fahrenheit is -1.48°F to 3.12°F range, it is totally sufficient for providing an estimate of atmospheric stability. This is an NRC accepted exception, per NRC SER Docket No. 50-302, "Conformance to Regulatory Guide 1.97", dated 6/16/87.

PARAMETER:

VARIABLE: PRIMARY COOLANT AND SUMP - GROSS ACTIVITY
TAG NO.: CA-54-CE
REF DWG: 302-700 Sht. 1

Type and Category - E, 3

Range - 1 microcurie/ml to 10 ci/ml

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 10, 11, 42

REASON:

To assess magnitude of radioactive releases. The Post Accident Sampling System (PASS) does not have a specific gross activity count monitor. The gamma spectrum counter is considered acceptable for this purpose.

Upgrade of PASS gives fully automatic sampling and measurement with a local indication in the count room.

PARAMETER:

VARIABLE: PRIMARY COOLANT AND SUMP - GAMMA SPECTRUM
TAG NO.: CA-54-CE
REF DWG: 302-700 Sht. 1

Type and Category - C, E, 3

Range - Isotopic Analysis

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 42

REASON:

To verify mitigation of RC system high radiation from breach of fuel cladding, and to assess magnitude of radioactive releases.

Upgrades of PASS gives fully automatic sampling and measurement with a local indication in the count room.

PARAMETER:

VARIABLE: PRIMARY COOLANT AND SUMP - BORON CONTENT
TAG NO.: CA-56-CE, CA-56-CI
REF DWG: 302-700 Sht. 2, 209-010

Type and Category - E, 3

Range - 0 - 6,000 ppm

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 42

REASON:

To assess magnitude of radioactive releases.

Duplicated previous item - RCS Soluble Boron Content (p.6)

Upgrade of PASS gives fully automatic sampling and measurement with a local indication in the count room.

PARAMETER:

VARIABLE: PRIMARY COOLANT AND SUMP - CHLORIDE CONTENT
TAG NO.: CA-57-CE
REF DWG: 302-700 Sht. 1

Type and Category - E, 3

Range - 0 - 20 ppm

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 35, 42

REASON:

To assess the magnitude of radioactive releases.

Upgrades of PASS gives fully automatic samplings and measurement with a local indication in the count room.

PARAMETER:

VARIABLE: PRIMARY COOLANT AND SUMP - DISSOLVED H_2 OR TOTAL GAS
TAG NO.: CA-55-CE and CA-55-CE2
REF DWG: 302-700 Sht. 2

Type and Category - E, 3

Range - 0 - 2,000 cc (STP) /KG

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 35, 42

REASON:

To assess the magnitude of radioactive releases.

Upgrades of PASS gives fully automatic samplings and measurement with a local indication in the count room.

PARAMETER:

VARIABLE: PRIMARY COOLANT AND SUMP - DISSOLVED OXYGEN
TAG NO.: N/A
REF DWG: N/A

Type and Category – N/A

Range - N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

0, 5, 42

REASON:

Ref: NRC Criteria Guidelines on NUREG-0737, Item II.B.3, Post Accident Sampling System, dated July 12, 1982

Criterion 4 of the reference stated that the measurement of oxygen is recommended but is not mandatory.

PARAMETER:

VARIABLE: PRIMARY COOLANT AND SUMP - pH
TAG NO.: CA-73-CE
REF DWG: 302-700 Sht. 2

Type and Category - E, 3

Range - 1 - 13

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 35, 42

REASON:

To assess the magnitude of radioactive releases.

Upgrades of PASS gives fully automatic samplings and measurement with a local indication in the count room.

PARAMETER:

VARIABLE: CONTAINMENT AIR - HYDROGEN CONTENT
TAG NO.: WS-10-CR, WS-11-CR
REF DWG: 205-062, WS-01, WS-02

Type and Category - E, 3

Range - 0 - 10%

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 42

REASON:

To assess the magnitude of radioactive releases.

Because of the installed redundant Hydrogen Monitors to meet Type C, Category 1 requirements (see p. 22), FPC has chosen to not install a grab sample capability for containment Hydrogen monitoring. Since the Hydrogen Monitors meet Type C, Category 1 requirements, they also meet Type E, Category 3 requirements.

PARAMETER:

VARIABLE: CONTAINMENT AIR - OXYGEN CONTENT
TAG NO.: N/A
REF DWG: N/A

Type and Category - N/A

Range - N/A

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - N/A

SOURCE:

0, 5, 42

REASON:

The NRC RG 1.97 required that Containment Oxygen be measured from 0 to 30% by volume. The category of the variable is 3 and the Type is E. A Type E variable is one that is "monitored as required for use in determining the magnitude of the release of radioactive materials, and for continuously assessing such releases." For a Type E variable, Category 3 items are considered as backup variables.

In discussions with the NRC, it was determined that the NRC expects the operator to compare the oxygen percentage with the hydrogen percentage to determine if the hydrogen formed is being caused by radiolysis or by metal-water reaction, which would be indicative of core damage.

Percentage of oxygen in the containment atmosphere is classified as a Type E variable. The definition of a Type E variable is that it is to be "monitored as required for use in determining the magnitude of the release of radioactive materials, and for continuously assessing such releases. However, the percentage of oxygen in the containment atmosphere does not provide the necessary information to determine the magnitude of releases of radioactive materials. At best, it provides a very indirect means of arriving at an order of magnitude estimate. There are other systems in place that can be used for this purpose. Some of these would be Containment Area Radiation, Radioactivity Concentration or Radiation Level in the Primary Coolant, Analysis of the Primary Coolant, Gross Activity and Gamma Spectrum of the Primary Coolant and Containment Sump, and Gamma Spectrum of the Containment Atmosphere. All of these systems provide a more direct means of determining the magnitude of the release and in addition most are Category 1 variables which means they are qualified to the same extent as a safety-related system.

The only other purpose of this variable then would be to allow the operator to determine what physical process is occurring that is forming the hydrogen in the Containment. Radiolysis occurs at all times, and is a slow process. It causes oxygen and hydrogen to be formed from water, so the percentages of both would increase providing no other processes were happening.

However, during a LOCA, a large amount of steam would be generated along with various other gases and the percentage of both hydrogen and oxygen would tend to be in a very dynamic state, rendering a reasonable decision based on that information virtually impossible.

A decrease in the percentage of oxygen along with an increase in hydrogen would be indicative of a metal-water reaction which in turn indicates core damage. Again, however, much better qualified instrumentation is available that provides a direct indication of core damage, rather than an indirect indication of core damage. Some of these systems are: Hot and cold Leg Water Temperatures, Core Exit Temperature, Coolant Inventory, Degrees of Subcooling, and the systems mentioned for determining the magnitude of the release. Additionally, the problems with a dynamic situation in the containment would also hold true in this case.

The requirement for providing the means of measuring containment oxygen content is not necessary because existing instrumentation provide more direct indication and are better qualified to perform the function of the required variable.

PARAMETER:

VARIABLE: CONTAINMENT AIR - GAMMA SPECTRUM
TAG NO.: WS-13-CE
REF DWG: 302-694

Type and Category - E, 3

Range - Isotopic Analysis

Environmental Qualification - N/A

Seismic Qualification - N/A

Quality Assurance - N/A

Redundancy - N/A

Power Source - N/A

Display - Local Panel

SOURCE:

0, 5, 42

REASON:

To assess the magnitude of radioactive releases.

Upgrade of PASS gives fully automatic sampling and measurement with a local indication in the count room.

SOURCE DOCUMENT

0. FPC letter #3F0884-07; 08/21/84
1. FPC letter #3F1185-17; 11/15/85
2. FPC letter #3F0386-11; 03/27/86
3. FPC letter #3F0687-09; 06/12/87
4. FPC letter #3F0188-03; 01/06/88
5. FPC letter #3F0388-18; 03/21/88
6. FPC letter #3F0688-06; 06/08/88
7. FPC letter #3F0988-06; 09/09/88
8. FPC letter #3F1089-26; 10/31/89
9. FPC letter #3F0190-06; 01/10/90
10. NRC letter #3N1085-12; 10/24/85
11. NRC letter #3N0687-12; 06/16/87
12. BWOG RG 1.97 Task Force
13. MAR 82-05-03
14. MAR 82-05-03-17
15. MAR 82-05-03-20
16. MAR 82-05-03-21
17. MAR 82-05-03-24 (Refuel 8)
18. MAR 83-11-14-01
19. MAR 84-08-10
20. MAR 84-08-10-02
21. MAR 84-08-10-04
22. SER 50-302; 01/13/84
23. SRP-6.2.4 - 6F, SRP-6.2.4 - 6J, NUREG-0737
24. FPC letter #3F1289-12; 12/15/89
25. MAR 80-11-17-03
26. MAR 89-10-23-01A
27. MAR 82-05-03-16
28. MAR 82-09-19-02
29. MAR 84-08-10-07
30. MAR 85-10-16-02
31. SER 50-302; 9/6/83
32. EQ 89-2613
33. MAR 83-03-04

34. MAR 89-10-16-01
35. MAR 90-06-19-01

SOURCE DOCUMENT (Cont')

36. NRC Letter #3N1293-26; 12/16/96
37. MAR 96-02-09-01
38. MAR 97-01-04-01
39. MAR 95-12-08-01
40. MAR 97-01-05-01
41. MAR 96-07-17-02
42. ES 96-07-17-00
43. REA 97-0265
44. MAR 96-03-12-01
45. FPC Letter #3F0797-21; 07/29/97
46. FPC Nuclear Licensing Memorandum, NL98-0005; 01/06/98
47. NRC Letter #3N1297-17; 12/23/97
48. MAR 82-05-03-05
49. MAR 82-05-03-06
50. MAR 82-05-03-07
51. MAR 82-05-03-12
52. MAR 96-07-17-01
53. MAR 99-04-01-01
54. ES 96-07-17-00, Reg. Guide 1.97 Instrumentation Study
55. License Amendment Request #234, Additional Post Accident Monitoring Instrumentation
56. MAR 96-07-17-06, "Core Flood Tank Level and Pressure Indicator Upgrade"
57. MAR 96-07-17-05
58. MAR 96-07-17-04
59. MAR 97-02-12-01, "High Pressure Injection Upgrade – Mechanical/Structural"
60. MAR 97-02-12-02, "High Pressure Injection Upgrade – Electrical/I&C"
61. Topical Design Basis Document for High Energy Line Breaks Inside Containment (Tab 9/5).
62. MAR 96-11-03-01, "Subcooling Margin Monitor Upgrade"
63. NRC Letter to FPC, Subject: CR-3 Staff Evaluation and Issuance of Amendment Regarding Subcooling Margin Monitoring Using SPDS (TAC No. MA4147), dated 4/20/99 (License Amendment No. 174)
64. MAR 97-07-05-01/02, "Control Room/Complex Emergency Ventilation System Improvement."

Florida Power
A Progress Energy Company

NUCLEAR OPERATIONS ENGINEERING
CRYSTAL RIVER UNIT 3

DESIGN BASIS DOCUMENT

FOR THE

SPDS/RECALL SYSTEM

SYSTEM CODE: EM
TAB: 5/12

ISSUE DATE: 1/25/88

	Revision 4	Revision 5	Revision 6	Revision 7	Revision 8	
Date:	4/26/99	6/28/99	2/9/00	9/20/01	12-13-01	
Design Engineer	R.P. Schmiedel	R.P. Schmiedel	R.P. Schmiedel	R.P. Schmiedel	R.P. Schmiedel	
Verification Engineer	K. L. Mansfield	P. A. Benyola	P. A. Benyola	L S. McGowan	L S. McGowan	
Supervisor	G. E. Englert	G. E. Englert	G. E. Englert	K. R. Wilson	K. R. Wilson	

REVISION LOG

Revision/Date	Description
0 - 1/25/88	Initial Issue - B&W Doc 51-1168648-00
1 - 3/19/93	This revision incorporated DBD T/C #52. Change bars not used due to the extensive nature of the changes. RECALL System removed from St. Petersburg.
2 - 8/01/97	Incorporated Temporary Change #601 as indicated by the change bars.
3 - 9/18/98	Incorporated Temporary Changes #625 and #789 as indicated by the change bars.
4 - 4/26/99	Incorporated Temporary Change #669 as indicated by the change bars.
5 - 6/28/99	Incorporated Temporary Change #730, which was a general re-write of the design basis document as a result of MAR 91-07-13-02 and calculation I84-0003, Revision 11. No Change bars were used.
6 - 2/9/00	Incorporated Temporary Change #1098 as indicated by the change bars.
7 – 9/20/01	Incorporated Temporary Change #1243 as indicated by change bars.
8 – 12-13-01	Incorporated Temporary Change #1254 as indicated by change bars.

DESIGN BASIS INDEX
FOR
SPDS/RECALL

Section Title **Page**

A. SUMMARY DESCRIPTION

The RECALL/Safety Parameter Display System (SPDS) was developed in response to the Nuclear Regulatory Commission (NRC) requirements (NUREG-0737 Supplement 1 and NUREG-0696) for improved man-machine interface during transient and abnormal plant operations. The system is designed to aid the control room operators in assessing the plant safety status, provide diagnostic type displays of plant parameters and environmental data to personnel manning the Technical Support Center (TSC), Emergency Operation Facility (EOF), and provide data link capabilities for other off-site locations.

OPERATIONAL FUNCTIONS

The purpose of the RECALL/Safety Parameter Display System (SPDS) is to assist the control room operator in the evaluation of plant conditions during both normal and abnormal situations. The system is available for use during conditions of plant heatup, cooldown, power operations, and abnormal plant conditions. Very abnormal conditions, such as Inadequate Core Cooling (ICC), are also displayed. The system is designed to use a minimum number of displays and parameters, yet it concisely presents to the operator information concerning the safety status of the following functions required by NUREG-0737, Supplement 1 and NUREG-0696:

a. Reactivity Control
b. Reactor Core Cooling and Primary System Heat Removal
c. Reactor Coolant System Integrity
d. Radioactivity Control
e. Containment Integrity

SAFETY FUNCTIONS

The RECALL/SPDS can be utilized to quickly focus on confined abnormal areas, thus limiting the number of control room indicators to be reviewed for further monitoring. The system was installed to meet the requirements of NUREG-0737, Supplement 1 and NUREG-0696.

RELEVANT DESIGN BASES FUNCTIONS

<u>Testing and Diagnosis</u>
The RECALL/SPDS is designed to provide for periodic testing to diagnose and recognize component degradation and system malfunctions.

<u>Environmental Qualification</u>
RECALL/SPDS is not required to meet single failure criteria and need not be qualified to meet Class 1E requirements. Thus the environmental qualification requirements of 10CFR50.49 do not apply. MAR 96-11-03-01 upgraded the SPDS to provide subcooling margin and meet the requirements of a RG 1.97, type A, category 1 variable. Most but not all of the RG 1.97 requirements are met as described in references 20, 21, and 22. Additional physical restraints are provided on major SPDS components to enhance seismic survivability.

Degrees of Subcooling (Digital Display)
The NRC issued License Amendment 174 on April 20, 1999 (reference 58a) approving revisions to the CR-3 Improved Technical Specifications (ITS) that the SPDS will be the primary display for subcooled margin. Previously the SPDS was a backup to the Tsat Indicators (RC-4-TI4 & 5). MAR 96-11-03-01, Subcooling Margin Monitor Upgrade (reference 10A) upgraded the SPDS to meet most, but not all, of the requirements to a Reg Guide 1.97 type A, category 1 parameter for subcooled margin. The exceptions are discussed in the FPC License Amendment request, reference 27a. Additionally, MAR 96-11-03-01 upgraded the 16 core exit thermocouple temperature loops to fully meet the requirements of Reg Guide 1.97 and installed two new recorders (RC-171-TR and RC-172-TR) on the main control board. These temperatures are provided to the SPDS for subcooling margin calculation. As a backup to the SPDS, subcooling margin can be determined by using Reg Guide 1.97 pressure and temperature instruments and plotting the data on instrument error corrected figures.

B. SYSTEM DESCRIPTION

GENERAL

The RECALL/SPDS is a redundant mini-computer based software system with multiplexers to access the analog and digital field input signals. The analog and digital input signals are non-nuclear-safety related.

The RECALL/SPDS computer uses the input data from the Plant Integrated Computer System (PICS) to provide indications on the video monitors of the RECALL/SPDS. The output of the PICS processor is transmitted on a local area network (LAN) for output on the main control board RECALL/SPDS displays through the RECALL/SPDS display computers installed behind the main control board. The control room operator has the ability to switch the control room displays between computers provided on each of the LANs. The hubs control the input to the LAN from the RECALL/SPDS processor. The hub will block signals from a processor from being transmitted on the LAN if the signal is detected to be erroneous.

The SPDS provides eight graphic displays, seven "ALERTS," and eight pages of alphanumeric displays to enable the operator to rapidly assess plant behavior under normal and abnormal conditions.

The eight graphic displays are:
1. Reactor Protection Trip Envelope Display
2. Power/Imbalance Trip Display
3. Power/Imbalance LCO Display
4. Wide Range P-T Limits-Normal MODE Display
5. Low Range P-T Limits-Normal MODE Display
6. Wide Range P-T Limits-EOP MODE Display
7. Low Range P-T Limits-EOP MODE Display
8. Inadequate Core Cooling Display

The seven alerts are:
1. Reactor Coolant System Inventory "RCS Inventory" ALERT
2. Reactivity "REACTIVE" ALERT
3. Reactor Building Conditions "RB Condition" ALERT
4. Reactor Building Pressure "RB Pressure" ALERT
5. Reactor Building Sump Level "RB Level" ALERT
6. "Radiation" ALERT

7. Decay Heat Removal System "DHRS" ALERT

The eight pages of alphanumeric displays which provide key parameters to the plant operator are grouped under the following headings:

1. Page 1 - Pressure and Valve Position Temperatures
2. Page 2 - Pressure and Valve Position Temperatures
3. Page 3 – Level
4. Page 4 – Flow
5. Page 5 - Reactor Building Conditions
6. Page 6 - Reactivity and Pump Status
7. Page 7 – Radiation
8. Page 8 - ES Trip Status

The RECALL/SPDS also provides specific plant information directly to the NRC. This set of data points is identified for use in the Emergency Response Data System (ERDS). A list of the CR3 computer points that make up this system is provided in Attachment 1.

C. MAJOR COMPONENT DESCRIPTION

The equipment used to generate the RECALL/SPDS display is located in the RECALL/SPDS display computer cabinet behind the main control board with the multiplexer control computer, which manages the data acquisition computers, located in the EFIC room. Located in the RECALL/SPDS display cabinet are RECALL/SPDS computers, video generators, a video monitor, and video amplifiers.

RECALL/SPDS displays are located on the main control board, in the Technical Support Center (TSC), and at the Emergency Offsite Facility (EOF). The display systems located on the main control board consist of 2 color monitors, and control panels for selecting desired displays. In the TSC there are three (3) computers that drive three big screen projection displays located on the wall of the conference room and there are two (2) computer workstations with display monitors. At the EOF there are four (4) work stations that display SPDS information.

The control panels for selecting displays are located directly below the monitors (see Figure 1). Displays can be changed by depressing the button for the desired display (Mode of Operation button). After the display comes up, selected curves can be added to the display by depressing the desired Curve Selection buttons.

* Selection of the RCS loop (A or B) that the display system is monitoring.
* Selection of primary or redundant instruments for T-Hot, T-Cold, and RCS pressure.
* Selection of Incore temperature rather than T-Hot on all displays except ICC. The ICC display always uses incore data for temperature display.
* Selection of curves or alpha-numeric displays.
* Selection of optional features that can be displayed on the selected curves such as HEATUP, NPSH, PTS, and Fuel Pin in Compression Curves and History Trace.
* An acknowledge function for an Alert condition.
* Printing the screen.

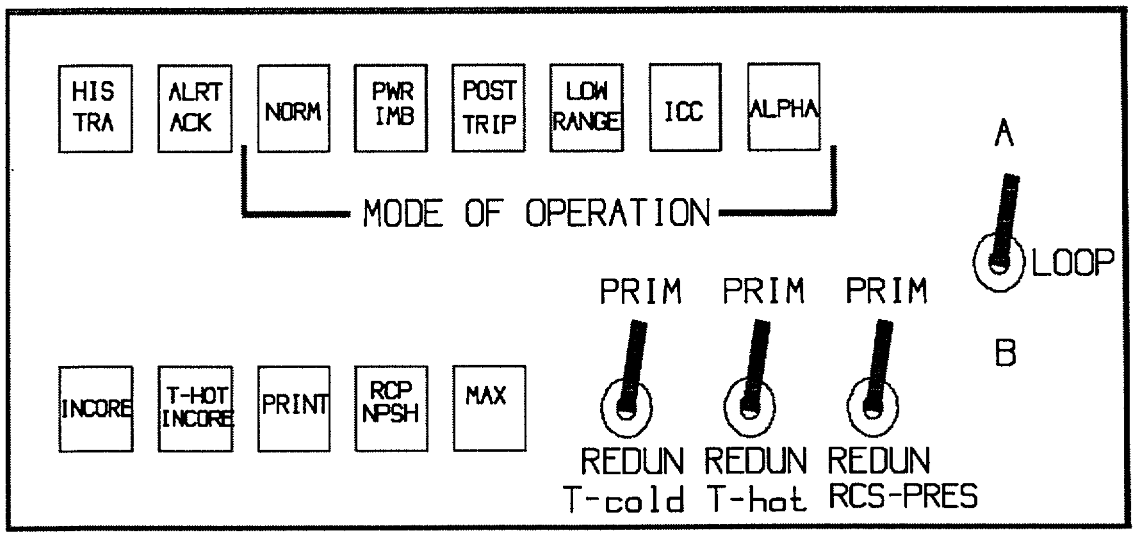

FIGURE 1: RECALL/SPDS CONTROL PANEL (SPDSPAN1)

Note: The primary/redundant switches are spring return to normal (center position).

A software system is utilized to allow for selection criteria at the remote RECALL/SPDS stations. A screen menu can be accessed through the use of hot keys associated with the individual computer at the remote station.

D. <u>SYSTEM OPERATION & CONTROL</u>

NORMAL OPERATION

System start-up is performed in accordance with OP-509, Safety Parameter Display System Operating Procedure. The sequence is fully automatic. The operator is only required to turn on the system components.

The system will go through a short self-diagnostics routine and display a test status. The RECALL/SPDS will then come up with the Post-Trip P-T curve displayed. If the reactor is not tripped, the History Trace will be off. If the reactor is tripped, the History Trace will be on.

Once the startup is complete, displays may be selected as desired. Should there be a loss of power to the system, it will automatically restart in the same sequence as a normal start up.

CONTROL PANEL OPERATION

Each display / display option is selected by a single push button. The push buttons are depressed and held momentarily in order for the selection to be completed. When a display is selected, there will be a listing of available curve selections across the bottom of the display. Whenever one of these curves is selected, the name of the selected curve, Heatup, Thermal Shock, etc., will be back-lit in Cyan and the appropriate curve will be added to the display.

All displays are provided with History Trace capability. When History Trace is selected by pressing the "HIS TRA" button on the control panel, the monitored parameters will leave a history trace on the display screen starting at that time and "History Trace" will be back-lit on the bottom of the screen.

History trace may be turned off at any time by pressing the "History Trace" button again. When history trace is turned off, the entire display will be cleared from the screen and returned without the history trace and the previous history trace will be lost. Also, whenever a display is changed all history traces will be lost.

REMOTE SOFTWARE CONTROL PANEL OPERATION

The use of hot keys controls the RECALL/SPDS display in the remote units. The following table provides the information for controlling these monitors.

Key	Function
A	Selects the Alpha Pages. There are 8 pages.
C	Selects or deselects the Incore measurement. This applies to certain displays only.
H	Enable or disable the History Trace feature. The history trace maintains the last 20 minutes of points, sampled at one minute intervals. When enabled, the plot points will remain until the history trace is disabled. if reselected, only the last 20 minutes are displayed.
Cntrl H	Control-H displays or Hides the button bar. The button bar mimics the function panel on the main control board. Displays or functions may be selected by using the mouse, depressing the underlined letter indicated on the button, or using function keys in the case of the selection of primary or redundant instruments or loop A and B.
I	Selects the Inadequate Core Cooling display.
K	Alert Acknowledge alerts indicate an out of tolerance condition. A flashing red alert indicates that the alert condition has occurred, is still present but has not been acknowledged by the operator. A solid red alert indicates that the alert condition has occurred, is still present but has been acknowledged by the operator. A flashing green alert indicates that an alert condition was present and has cleared, i.e. returned to normal condition, but was not acknowledged.
L	Selects the Low Range display.
M	Maximize the RECALL/SPDS display window. If maximized, the window is restored to its original size.
N	Selects the Norm display.
P	Selects the Power Imbalance display. When first selected, the LCO curves are displayed. Depressing P again selects the Trip curves.
R	Selects pump combinations for RCP-NPSH. Repeated depression pages through single, 2/0, 0/2, and back to normal.
T	Selects the Post Trip display.

The following keys perform the indicated functions:	
Shift-F3	T-Cold Primary
Shift-F4	T-Cold Redundant
Shift-F5	T-Hot Primary
Shift-F6	T-Hot Redundant
Shift-F7	RCS Pressure Primary
Shift-F8	RCS Pressure Redundant
Shift-F9	Loop A
Shift-F10	Loop B

DISPLAY DESCRIPTIONS

All displays follow a color code scheme to enhance the displays and improve readability. Colors on the RECALL/SPDS have the following associations:

Black - Screen Background Color; reverse video letters.
Red - T-hot; also for parameter in alarm/alert condition.
Cyan - T-cold; also to indicate curve selection.(light blue)
Yellow - Primary System Parameters.
Green - Secondary System Parameters.
White - Background drawings and variable names; reverse video background.
Magenta - T-incore
Blue - Not used at present time

Bar Chart displays for the OTSG levels and the Pressurizer level will go RED when the indicated values exceed their specified Tolerance Limits. This will aid in rapid recognition of possible problems which may need immediate actions to correct.

ABNORMAL OPERATION

RECALL/SPDS contains a feature to alert the operator of potential problems in the control of those safety functions which are not rigorously monitored on the available displays. Seven "ALERT" conditions are listed in the upper right corner of all RECALL/SPDS screens. In the event that an abnormal event occurs in one of the systems monitored by the "ALERT" feature, that "ALERT" message will flash red until the operator acknowledges the condition by pressing the alert acknowledge, "ALRT ACK," button. At that time, the message will go solid (i.e. continuous illumination). When the alert condition clears, the message display will automatically clear and the logic will be reset and again be available for actuation.

The system is programmed to display and monitor the 7 alert status conditions. They are:

1. <u>RCS INVENTORY</u>
The purpose of the RCS INV alert is to monitor the Reactor Coolant System Level and warn the operator of a parameter exceeding a limit. With RCPs not running, Reactor Vessel Level (RVL) and Hot Leg Level (HLL) are the parameters monitored (via Reactor Coolant Inventory Tracking System (RCITS)).

2. <u>REACTIVITY</u>
The Reactivity Alert will be actuated if all rods are not inserted and 3 seconds have elapsed since the reactor trip signal was received or, the source range counts are > 1000 CPS 25 minutes after a reactor trip confirm has been received.

3. <u>RB CONDITIONS</u>
The purpose of the "RB CONDITION" alert is to monitor the parameters that can cause a change in the RB integrity that are not rigorously monitored elsewhere within RECALL/SPDS. The parameters monitored by this alert are average RB Temperature (127.25°F), RB Hydrogen (>3.5 %), RB radiation (>6R/Hr), and ES (RB ISOL) actuation. RB temperature will not cause the alert until the average of all four of the above signals equal or exceeds 127.25°F.

4. <u>RB PRESSURE</u>
The RB PRESS alert will warn the operator of increasing pressure conditions in the RB before reaching the ESAS actuation setpoint. The RB pressure ALERT will alarm and be displayed anytime the narrow range RB pressure signal is less than –1.78 psig or greater than +2. 78 psig. Redundant high range RB pressure signals are displayed on Alpha Page 5 for operator checking of this ALERT.

5. <u>RB LEVEL</u>
The purpose of the "RB LEVEL" Alert is to warn the control room operator that a level above 4 feet exists in the RB Sump.

6. <u>RADIATION</u>
The Radiation Alert will be actuated to warn the control room operator of increasing radiation levels in one or more of the monitored release paths.

7. <u>DHRS</u>
The DHRS Alert is responsive during the Decay Heat mode of RCS operation only. This ALERT is activated when RCS pressure is less than 284 psig and T-cold in both loops is less than 300°F.

   After these conditions have been met, the alert then starts by checking for proper pressurizer level based upon RCS T-cold Temperature. If both RCS T-cold A & B are below 283°F, and Pressurizer level is greater than or equal to 220 inches, the alert is set.

   Decay Heat Pump (DHP) status is checked next. If DHP A or B is running, flow is verified to be less than or equal to 2800 gpm, or greater than or equal to 4000 gpm. If it is not, the ALERT will be set. If a pump is not running, it will not set the ALERT for that pump due to flow.

PARAMETER	REFERENCE/ SOURCE	REASON
DESIGN BASIS FOR SPD/RECALL:		
The safety parameter display system (SPDS) shall be located convenient to the control room operators.	2	Display must be available to plant operating personnel where corrective actions can be taken.
The SPDS should provide a concise display of critical plant variables.	2, 18, 38	Display should be concise in order to reduce interpretation and promote rapid assessment.
The SPDS will continuously display information from which the plant safety status can be readily and reliably assessed.	2, 18, 38	To provide real time plant conditions.
The SPDS shall be in operation during normal and abnormal operating conditions and be capable of displaying pertinent information during steady state and transient conditions. SPDS is qualified to meet the requirements of RG 1.97, Type A, Category I, with exceptions.	1, 10a, 18, 27a 38	Need for determining plant safety status can arise during any operating mode and may be needed most during abnormal or critical operating conditions. SPDS display of subcooling margin is used for transient mitigation.
The SPDS shall be capable of presenting the magnitudes and the trends of parameters or derived variables. The display parameter trending display shall contain recent and current magnitudes of the parameter as a function of time.	1, 38, 59	To better assess current plant conditions and reduce reaction time for corrective action by anticipating approaching parameter limits.
All data for display shall be validated where practicable on a real-time basis as part of the display to control room personnel. For example, redundant sensor data may be compared, the range of a parameter may be compared with predetermined limits, or other quantitative methods may be used to compare values. When an unsuccessful validation of data occurs, the SPDS shall contain means of identifying the impacted parameter(s).	1, 59	To ensure that the SPDS presents the most current and accurate status of the plant and is not compromised by unidentified faulty processing or failed sensors. When the redundant instrument comparison is made and limits are exceeded an "R" is displayed in red next to the parameter. If a validity limit is exceeded an "X" is displayed next to the parameter.
The SPDS shall be designed such that no operating personnel other than the normal control room operating staff are required for its operation.	1, 59	Rapid and accurate assessment of plant safety status requires that SPDS displays be controlled from within the control room.

PARAMETER	REFERENCE/ SOURCE	REASON
The SPDS shall be designed to incorporate accepted human factors principles so that the displayed information can be readily received and comprehended by SPDS users.	2, 18, 38	So that the displayed information can be readily perceived and comprehended by the user and, under accident conditions, aid in executing the system oriented emergency procedures with minimum interpretation and movement.
The SPDS shall be suitably isolated from electrical or electronic interference with equipment and sensors that are in use for safety systems.	2, 38	The SPDS must not degrade the capability of a safety system.
The SPDS need not meet requirement of the single failure criteria and it need not be qualified to meet Class 1E requirements. The SPDS need not be seismically qualified. Most, but not all of the RG 1.97 requirements for Type A, Category I variables.	2, 10a, 27a, 58a	The SPDS is used in addition to the basic components for safe reactor operation and serves to aid and augment these components. Thus, requirements applicable to control room instrumentation are not needed for this augmentation. The SPDS display of subcooled margin is required to meet the requirements of RG 1.97, with exception, (Refs. 27a and 58a).
Design provisions shall be included in the interfaces between the SPDS and non-safety systems to ensure the integrity of the SPDS upon failure of non-safety equipment.	1, 20	Failure of non-safety related equipment must not disable SPDS.
The SPDS shall be designed to provide for periodic testing to diagnose and recognize component degradation and systems malfunction.	1, 59	Detection of system failures.
The SPDS, as used in the control room, shall be designed to an operational unavailability goal of 0.01 cold shutdown and refueling modes for the reactor shall be 0.2.	1, 26	To demonstrate system reliability.
The Licensee shall prepare a written safety analysis describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents.	2, 18, 38	To document the basis for the parameters used to assess the safety status of each identified function for a wide range of events.
The SPDS shall be designed to operate satisfactorily in the expected environment in which it is located.	60	Provide for operation in the expected environment.

PARAMETER	REFERENCE/ SOURCE	REASON
The minimum information to be provided shall be sufficient to provide information to plant operators about: • Reactor control • Reactor core cooling and heat removal from the primary system • Reactor Coolant System integrity • Radioactivity control • Containment Conditions	2, 18, 19, 38	These are the minimum functions to assess plant safety status.
The SPDS provides three major types of information (A, B and C below):	19, 21	Each graphic display shows plant operating parameters and parameter limits which should bound operation for a specific plant mode or anticipated event. These displays compliment the Emergency Operating Procedures Technical Basis Document (Reference 23).
A. EIGHT GRAPHIC DISPLAYS 1) Reactor Protection Trip Envelope Display 2) Power/Imbalance Trip Display 3) Power/Imbalance Limit Display 4) Wide Range P-T Limits-Normal MODE Display 5) Low Range P-T Limits-Normal MODE Display 6) Wide Range P-T Limits-EOP MODE Display 7) Low Range R P-T Limits-EOP MODE Display 8) Inadequate Core Cooling Display These displays are further described in the following pages.		

PARAMETER	REFERENCE/ SOURCE	REASON
The eight graphic displays are developed from the following plant parameter signals:	6, 18, 19, 28 through 32, 38	These variables are necessary to implement the eight graphic displays for the SPDS Redundant signals are compared by the SPDS and discrepancies (durations greater than five times the string inaccuracy) are identified on the CRT by a question mark "?" next to the data.
RANGE	**Variable Range**	
a) WR T-cold A, primary and backup signals. b) WR T-cold B, primary and backup signals.	50 – 650°F	Plotted and displayed with RCS pressure on P-T displays to monitor plant status against NDT, thermal shock, fuel compression and RCP-NPSH limits, and to provide symptoms of excessive or inadequate heat transfer when used with OTSG saturation temperature.
c) WR T-hot, primary and backup signal d) WR T-hot, primary and backup signal	120 – 920°F	See discussion of T-cold A&B. In addition, it is displayed against subcooling margin and saturation limits for symptoms of inadequate core cooling. It is also used with incore thermocouple temperature as an indication of natural circulation.
e) WR RCS Pressure A, primary and backup signal. f) WR RCS Pressure B, primary and backup signal.	0-2500 psig (primary) 0-3000 psig (backup)	See discussion of T-cold A&B. Also plotted with incore thermocouple temperature to monitor natural circulation or superheated conditions during ICC.
g) LR RCS Pressure A h) LR RCS Pressure B	0-600 psig	Used as RCS pressure signal during low RCS pressure operation.
i) Incore thermocouples temperature, highest of 8 qualified input signals for the selected loop.	0-2500°F	Average value plotted with RCS pressure on ICC curve during ICC conditions and can be plotted with RCS pressure on other P-T curves. (See discussion of RCS pressure, WR).

PARAMETER	REFERENCE/ SOURCE	REASON
j) NI Power, highest of four (4) input signals.	0 – 125%	Used in conjunction with delta flux to indicate percent of rated power on the Power/Imbalance display. Also used to indicate percent P-T. All four (4) signals shown under REACTIVITY on Alphanumeric "Page 6."
k) Imbalance (Delta Flux) average of four (4) input signals.	-62.5 to 62.5%FP	Used in conjunction with NI power to indicate power imbalance on Power/Imbalance display.
l) RCS Flow A m) RCS Flow B	0-80E6 lb/hr	Expressed as percent of total RCS flow, indicated on Trip Envelope, Imbalance, LR P-T and ICC.
n) OTSG Steam Pressure A o) OTSG Steam Pressure B	0-1200 psig	Used as input to OTSG saturation temperature on Post-Trip, LR P-T, and ICC displays. Used to display symptoms of excessive heat transfer and inadequate heat transfer when used with T-cold.
p) OTSG Startup Level A q) OTSG Startup Level B	0-250 inches	See discussion on OTSG Op. Range A & B. SU Range may be useful in OTSG tube rupture identification, heatup and cooldown operations, and OTSG drain/fill operations.
r) OTSG Operate Level A s) OTSG Operate Level B	0-100%	Displayed on all graphic displays primarily for immediate detection of excessive feedwater and loss of feedwater events.

PARAMETER	REFERENCE/ SOURCE	REASON
GRAPHIC DISPLAYS:		
1. Reactor Protection Trip Envelope display.	6, 12, 16, 19	The Reactor Protection Trip Envelope P-T display provides the RPS trip setpoints from the CR3 Technical Specifications and a normal operations box. This display is available on one of two CRTs during normal operations and provides the control room operator with indication of normal and off-normal conditions to allow anticipation of a reactor trip. On this display reactor coolant temperature is displayed on the horizontal axis with a range from 520°F to 620°F while reactor coolant pressure displayed on vertical axis with a range from 1700 to 2500 psig. These ranges provide sufficient resolution of reactor coolant conditions within normal conditions and make conditions challenging protection system setpoints readily discernible. Numerical indication of % RCS flow and % RX Power are also displayed, as well as bar charts and numerical values for OTSG Startup Level, Operate Level, and Pressurizer Level.

PARAMETER	REFERENCE/ SOURCE	REASON
2. Power/Imbalance Trip Display	12, 19	Certain accidents involving high reactivity rates occur too rapidly to permit effective protection actions based upon either reactor coolant temperature or pressure. There are thermal limits associated with these accidents expressed in terms of linear power peaking (kw/ft) or Departure from Nucleate Boiling Ratio (DNBR) limits, but are physically monitored in terms of neutron flux, both absolute level (% rated thermal power) and distribution (imbalance reactor flux in top half of core minus that in lower half, also expressed as % rated thermal power). These limits are modified to reflect the number of operating reactor coolant pumps (RCPs). These limits are fuel cycle dependent. Both reactor trip and LCO limits are provided. Reactor power imbalance ranging from –50% to +50%, is left to right along the horizontal axis and reactor power as percent scaled thermal power, increasing from 0 to 110% bottom to top is along the vertical axis. Only the limits for 4 and 3 RCPs, the allowable RCP combinations at power, are provided. Numerical indication of % RCS flow and % RX power are also displayed, as well as bar charts and numerical values for OTSG Startup Level, Operate Level and Pressurizer Level.
3. Power/Imbalance Limit Display	12	Same as Power/Imbalance above; however this curve is based on LCO limits.

PARAMETER	REFERENCE/ SOURCE	REASON
4. Wide Range P-T Limits-Normal MODE Display	12, 15, 19, 28 through 32	The displays consist of permanent and selectable curves shown on a pressure versus temperature format. The curves are consistent with CR3's Technical Specifications and operating procedures. Numerical indication of GPM EFW flow, bar charts, and numerical values for OTSG Startup Level, Operate Level, and Pressurizer Level are displayed. When reactor trip occurs, the Post-Trip display, with history trace on, is automatically selected.
Permanent curves consist of: Axis for the Post-Trip P-T	12, 19	The horizontal axis represents temperature in degrees Fahrenheit with a range from 250 to 650°F. This temperature may be T-hot from Loop A or B, T-cold from Loop A or B, the highest incore thermocouple for the selected loop. One CRT can be used to show either loop's information but not both loops at the same time. Saturation secondary steam temperature is another variable that is displayed on the horizontal axis. The vertical axis represents pressure in PSIG. This pressure may be RC pressure or OTSG steam pressure with a range from 0 to 2500 psig.
Post Trip Window	12, 19	The "post trip" operating window is drawn to show where the reactor coolant pressure and temperature should end up after reactor and turbine trip. It is possible to end slightly outside the window and still have a stable plant; however, this window gives a "first" basis for determining if the plant is operating "normally" after a trip.
Post Trip T-Hot and T-Cold "Target" Boxes	12	These boxes within the post trip window represent the normal temperature range at 100% power. This is a "target" box rather than an operative requirement.

PARAMETER	REFERENCE/ SOURCE	REASON
Saturation Curve	12, 19	The saturation curve is a permanent curve. The data points are taken from the ASME Steam Tables 1967 and changed to PSIG.
Subcooled Margin Curve	12,13, 19, 28 through 32	This curve is the error adjusted saturation curve plotted to the left of the saturation curve. The curve starting point is at (197, 67), corresponding to the instrument error corrected value for 212°F and 0 psig. Below this temperature there is no concern for subcooling margin, thus, the curve is drawn straight down to the horizontal axis at (197, 0).
Saturation Temperature of Secondary System	12, 19	A vertical line is displayed on the P-T display that represents the saturation temperature of the secondary system (Tsat secondary) based on secondary steam pressure for the loop. The relative movement of Tsat secondary to the RCS cold leg conditions is the means by which one key ATOG symptom of primary-to-secondary heat transfer is evaluated.
Cooldown Curve	12, 19	The cooldown Nil Ductility Temperature (NDT) limits on limiting RCS components for the first 32 effective full power years are shown on the cooldown curve. The curve may be modified after that time.
Selectable Curves	12, 19	The Post-Trip display (Normal P-T Display) has a number of control room operator selectable curves associated with it to enhance its usefulness during normal and abnormal conditions. There are selectable curves for RCP Net Positive Suction Head (NPSH) for pump combinations 2/0, 0/2, 1/0, 0/1 and 0/0.
RCP-NPSH Limits	12, 19	During plant cooldown, the operator needs to know the limits for the Reactor Coolant Pumps based on NPSH. There are six curves provided; the 0/2, 2/0 for the 'A' and 'B' loop, 0/1, 1/0 for either loop, and Natural Circulation Cooldown.

PARAMETER	REFERENCE/ SOURCE	REASON
5. Low Range P-T Limits-Normal MODE Display	12, 19	Same as Wide Range P-T Limits-Normal Display, except that pressure axis is from 0 to 650 psig and temperature axis is from 50 to 475°F
6. Wide Range P-T Limits-EOP MODE Display	12, 19	Same as Wide Range P-T Limits-Normal Mode Display, except for EOP Mode, no curve for Natural Circulation Cooldown, OTSG Levels switch to EFIC levels for input and the addition of a Superheat curve.
<u>Superheat Curve</u>	12, 19	The Superheat curve is a permanent curve on the Wide Range and Low Range – EOP Mode displays. This curve is an error adjusted curve derived such that Tsat + Instrument error = Superheat Limit.
7. LOW RANGE P-T Limits-EOP MODE Display	12, 19	Same as normal display above, except that there is no curve for Natural Circulation Cooldown.
8. Inadequate Core Cooling Display	12, 19	The inadequate core cooling display is a series of curves that will provide operator actions based on the Emergency Operating Procedures Technical Basis Document (reference 23), and Operating Guidelines for Small Breaks for CR-3 (reference 22). The curves consist of the saturation line which provides reference point from the post trip P-T curve to the ICC curves; the subcooling margin monitoring curve, and the margin to superheat curve, a 1400°F clad temperature curve; and an 1800°F clad temperature curve. These curves break the display into four (4) distinct regions. These regions are identified on the screen. Specific operator actions will be required for each of these regions.

PARAMETER	REFERENCE/ SOURCE	REASON
B. ALERTS	12, 19, 28 through 32	The Alerts on the CR3 SPDS are to warn the control room operator of potential problems in the control of those safety functions which are not being rigorously monitored on the eight available displays. When conditions warrant, the alert message will appear on the CRT screen and will flash until the control room operator acknowledges the alert signal. The alert message will remain on the screen until the conditions do not warrant an alert. If an alert clears before the operator has acknowledged it, the alert will flash (green background with white text and alternated white background with green text). The alert message will clear when acknowledged.

ALERTS:

There are seven alerts as follows:

PARAMETER	REFERENCE/ SOURCE	REASON
1. Reactor Coolant System Inventory "RCS Inventory" Alert Signals Monitored: Reactor Vessel Level A Reactor Vessel Level B Hot Leg Level A Hot Leg Level B Void Fraction A Void Fraction B Reactor Coolant Pump Status A, B, C, D	12, 19, 41	The "RCS Inventory" alert is to monitor the Reactor Coolant System level and warn the operators of a parameter exceeding a limit. This alert is activated if one of the following occurs: 1. With all RCPs not running, either Reactor Vessel Level (RVL) falls below 18.3% or either Hot Leg Level (HLL) falls below 40.2%. 2. With any RCP running, Void Fraction is monitored. Void Fraction does not have a limit specified and does not initiate an alert.

PARAMETER	REFERENCE/ SOURCE	REASON
2. Reactivity "REACTIVE" Alert Signals Monitored: Source Range NI-01 Source Range NI-02 All Rods in Reactor Tripped	12, 19, 28 through 32	The "REACTIVE" alert warns the control room operator that adequate control of the reactivity safety function is not occurring. This alert is activated if either one of the following occurs: 1. All control rods are not fully inserted within three (3) seconds following a reactor trip or an asymmetric fault exists. 2. The neutron flux has not decayed to rate at or below 10^3 cps within 25 minutes following a reactor trip.
3. Reactor Building Conditions "RB Condition" Alert Signals Monitored: RB Temp - 102 ft 9 inches - 125 ft - 180 ft 6 inches - 235 ft RB H2 Concentration A & B RM-G29, RM-G30, RM- A1, RM-A6 18 ES Actuation Channels Group 1: Ch RC1 A, Ch RC2 A, Ch RC3 A Group 2: Ch RC4 A, Ch RC5 A, Ch RC6 A Group 3: Ch RB1 A, Ch RB2 A, Ch RB3 A Group 4: Ch RC1 B, Ch RC2 B, Ch RC3 B Group 5: Ch RC4 B, Ch RC5 B, Ch RC6 B Group 6: Ch RB1 B, Ch RB2 B, Ch RB3 B	12, 19, 41	The "RB Condition" alert is to monitor the parameters that can cause a change in RB integrity, and warn the operators if any RB parameter exceeds its limits. The alert is activated if one of more of the following occur: 1. The average of all RB Temperature signal is $\geq 127.25°F$. 2. RB Hydrogen Concentration A or B exceeds 3.5%. 3. Any one of the RB radiation monitors exceed their setpoint. RM-G29 or RM-G30 >6 R/HR RM-A1 or RM-A6 5000 CPM 4. During an ES actuation the message "RB ISOL SEE ES LIGHT MATRIX" is shown. The "RB ISOL" informs control room operator that signals for the Engineered Safeguards Actuation System to actuate are present. There are a total of 18 ES channels monitored, arranged in six groups of three. When two of three redundant channels in any of these six groups indicates "tripped," then the "RB ISOL" alert will be alarmed and displayed.

PARAMETER	REFERENCE/ SOURCE	REASON
4. Reactor Building Pressure "RB Pressure" Alert Signals Monitored: RB Pressure Narrow Range Signal	12, 19, 28 through 32	The "RB Pressure" alert warns the control room operator of increasing pressure conditions in the Reactor Building before they reach ESAS actuation setpoint. The "RB Pressure" alert will alarm and be displayed any time the condition exists where the RB pressure narrow range signal is <-1.78 psig or >+2.78 psig.
Reactor Building Sump Level "RB LEVEL" Alert Signals Monitored: RB Sump Level A & B	12, 19, 41	The "RB LEVEL" alert warns the operator when the RB Sump level exceeds 4 feet.

PARAMETER	REFERENCE/ SOURCE	REASON
6. RADIATION" Alert	12, 19, 28 through 32	The "RADIATION" alert warns the control room operator of increasing radiation levels in one or more of the monitored release paths. The "RADIATION" alert will alarm and be displayed any time the condition exists where one or more radiation monitors exceed its alarm setpoint.
Signals Monitored:	Setpoint	
a. RM-G29 RB Dome	6R/HR	a) Help discriminate between major LOCA and Steam Line Break.
b. RM-A1 RB Purge	5K CPM	b) Provides early indication of vent system radioactivity possible unplanned release.
c. RM-A2 FH Duct	1K CPM	c) Same as b.
d. RM-A6 RB Vent	5 CPM	d) Provides early indication of RCS leakage.
e. Vacuum Pump RM-A12 Condenser	500 CPM	e) Provides indication of steam generator tube leakage.
f. RM-G25 Steam Line	1 MR/HR	f) Will identify affected steam generator when a tube leak is present.
g. RM-G26 Steam Line	50 GPD	g) Same as f.
h. RM-G27 Steam Line	50 GPD	h) Same as f.
i. RM-G28 Steam Line	1 MR/HR	i) Same as f.
j. RM-L1 Primary Coolant	80K CPM	j) Provides early indication of fuel cladding failures.
k. RM-L2 Plant Discharge	600K CPM	k) Provides early indication of abnormalities in the liquid effluent discharge flow path.
l. RM-L7 Turbine Building Sump	5K CPM	l) Same as k.
m. RM-A5 Control Room Vent	200 CPM	m) Provides indication of degrading control room environment prior to actuations occurring.
n. RM-G1 Control Room	2 MR/HR	n) Same as m.
o. RM-G30 RB Dome	6R/HR	o) Same as a.

PARAMETER	REFERENCE/ SOURCE	REASON
7. Decay Heat Removal System "DHRS" Alert Signals Monitored: RCS Pressure LR A & B T-cold A & B (Selected Primary or Redundant) Pressurizer Level Decay Heat Pump A&B Run Status Decay Heat Pump A&B Flow	10, 12, 27	For the DHRS alert to be set, four conditions are checked. The first must be true and at least one of the others must be true for the alert to be set. 1. RCS pressure < 284 psig in either Loop A or B, AND T_{Cold} < 300°F, in both loops. 2. T_{Cold} < 283°F in both loops, AND PZR LVL=220 inches. 3. DHP A is running, AND Loop A flow <=2800 gpm OR >=4000 gpm. 4. DHP B is running AND Loop B flow <=2800 gpm OR >=4000 gpm

PARAMETER	REFERENCE/ SOURCE	REASON
ALPHANUMERIC DISPLAY: C. ALPHANUMERIC DISPLAY	12, 19, 28 through 32	The Alphanumeric Display is an eight page display. Page 1 provides key primary and secondary PRESSURE information and the pressurizer valve status as well as Pressurizer relief VALVE POSITION information. Page 2 shows Temperature Information (e.g., T-hot, T-cold, T-incore, RCS, etc). Page 3 provides key LEVEL information for primary and secondary systems. Page 4 provides system FLOW information (e.g., RCS flow, HPI flow, MFW flow, and EFW flow). Page 5 provides information on the parameters monitored for the Reactor Building. RB CONDITIONS include the RB Sump Level, RB Flood Level, RB High Rad, RB Purge Exhaust, Air Sample, Hydrogen Concentration, RB Temperature, and Pressure. Page 6 provides displays for Reactivity (e.g., NI detectors) and Pump Status (e.g., RC pump and DH/LPI Pump status). Page 7 provides specific status information on the fifteen (15) radiation monitors which are used in the RADIATION Alert. Page 8 provides information concerning each ESAS channel trip status. This information is used in the RB Conditions Alert to warn the operator of the RB Isolation condition and then sends the operator to the ES MATRIX to determine the RB Isolation valve status. If an alert message is received, parameter information is question marked, or an abnormal situation occurs, the control room operator can select the appropriate page of the Alphanumeric display to

PARAMETER	REFERENCE/ SOURCE	REASON	
		determine which parameter(s) is (are) out of tolerance.	
Slave Tsat Indicators, EMCO-66 and 67	10a, 27a, 8a	Tsat Indicators EMCO-66 and 67 are driven by the SPDS and indicate the SPDS calculated value of subcooling margin as displayed on the SPDS monitors. These indicators provide readability at a greater distance from the Control Board than does the SPDS display alone.	
	SIGNALS MONITORED (per page)		
PAGE 1			
PRESSURE			
RCS Pressure Wide Range A Loop	RC-3A-PT3	RECL-4	
RCS Pressure Wide Range (A Loop)	RC-158-PT	RECL-224	
RCS Pressure Wide Range B Loop	RC-3B-PT3	RECL-5	
RCS Pressure Wide Range (B Loop)	RC-159-PT	RECL-225	
RCS Loop A Pressure LR	RC-147-PT	RECL-243	
RCS Loop B Pressure LR	RC-148-PT	RECL-40	
OTSG A Outlet Steam Pressure	SP-6A-PT1/2	RECL-104	
OTSG B Outlet Steam Pressure	SP-6B-PT1/2	RECL-105	
Nuclear Service CCC Disch. Header Pressure	SW-2-PT	RECL-237	
VALVE POSITION			
PZR Code Safety Valve RCV-8	RC-160-ME1A	RECL-176	
PZR Code Safety Valve RCV-9	RC-160-ME2A	RECL-177	
PORV Position RCV-10	RC-160-ME3A	RECL-175	
PAGE 2			
TEMPERATURE			
T-hot Wide Range A Loop	RC-4A-TE1	RECL-17	
T-hot Loop A Wide Range	RC-4A-TE4	RECL-239	
T-hot Wide Range B Loop	RC-4B-TE1	RECL-18	
T-hot Loop B Wide Range	RC-4B-TE4	RECL-240	
T-cold Wide Range A Loop	RC-5A-TE2	RECL-7	

PARAMETER	REFERENCE/ SOURCE	REASON
T-cold Loop A Wide Range	RC-5A-TE4	RECL-241
T-cold Wide Range B Loop	RC-5B-TE2	RECL-8
T-cold Loop B Wide Range	RC-5B-TE4	RECL-242
T-incore Ave Temp	Ave of 16 qualified CETs	Calculated in SPDS RECL-419
T-incore Ave Temp A Train		RECL-421
T-incore Ave Temp B Train		RECL-422
T-incore Max Temp A Train		RECL-423
T-incore Max Temp B Train		RECL-424
RCS Cooldown Rate (1 min Ave)		RECL-427
RCS Cooldown Rate (10 min Ave)		RECL-428
RCS A Delta Temperature		Calculated in SPDS
RCS B Delta Temperature		RECL-411
		RECL-412
Code Safety Tailpipe Temperature - RCV-8	RC-17-TE1	Calculated in SPDS
Code Safety Tailpipe Temperature - RCV-9	RC-17-TE2	RECL-126
PORV Tailpipe Temperature	RC-17-TE3	RECL-125
		RECL-124

PAGE 3

LEVEL

Pressurizer Level (Compensated)	RC-1-LT1/3	RECL-66
Makeup Tank Level	MU-14-LT1/2	RECL-50
Hot Leg A Level	RC-163A-LT1	RECL-63
Hot Leg B Level	RC-163B-LT1	RECL-70
Reactor Vessel A Level	RC-164A-LT1	RECL-62
Reactor Vessel B Level	RC-164B-LT1	RECL-65
RC Void Trend (A Loop)	RC-169A-JY	RECL-64
RC Void Trend (B Loop)	RC-169B-JY	RECL-71
Emergency Feedwater Tank Level	EF-98-LT	RECL-236
OTSG A Startup Level	SP-1A-LT4/5	RECL-90
OTSG B Startup Level	SP-1B-LT4/5	RECL-91
OTSG A Operating Level	SP-1A-LT2/3	RECL-88
OTSG B Operating Level	SP-1B-LT2/3	RECL-89
OTSG A Full Range Level	SP-1A-LT1	RECL-92
OTSG B Full Range Level	SP-1B-LT1	RECL-93

PARAMETER	REFERENCE/ SOURCE		REASON
OTSG A Low Level	EFIC avg < 12.5"	RECL-432	
OTSG B Low Level	EFIC avg < 12.5"	RECL-433	
OTSG A Hi Level	EFIC avg > 93.6%	RECL-434	
OTSG B Hi Level	EFIC avg > 93.6%	RECL-435	
EDG 3A KW	EG-41-JY	RECL-133	
EDG 3B KW	EG-42-JY	RECL-134	

PAGE 4

FLOW

RCS Flow A Loop	RC-14A-FT	RECL-31	
RCS Flow B Loop	RC-14B-FT	RECL-32	
Letdown Flow	MU-4-dPT	RECL-122	
RC Makeup Flow	MU-24-dPT	RECL-123	
HPI WR Flow (Loop B1) MUV-25	MU-23-dPT1	RECL-51	
HPI WR Flow (Loop A2) MUV-23	MU-23-dPT2	RECL-52	
HPI WR Flow (Loop B2) MUV-26	MU-23-dPT3	RECL-53	
HPI WR Flow (Loop A1) MUV-24	MU-23-dPT4	RECL-54	
HPI LR Flow (Loop B1) MUV-25	MU-23- dPT9	RECL-260	
HPI LR Flow (Loop A2) MUV-23	MU-23- dPT10	RECL-261	
HPI LR Flow (Loop B2) MUV-26	MU-23- dPT11	RECL-262	
HPI LR Flow (Loop A1) MUV-24	MU-23- dPT12	RECL-263	
DH LPI Flow A Pump GPM	DH-1-FT3	RECL-55	
DH LPI Flow B Pump GPM	DH-1-FT4	RECL-56	
Main Feedwater Flow A OTSG	SP-8A-dPT1/dPT2	RECL-100	
Main Feedwater Flow B OTSG	SP-8B-dPT1/dPT2	RECL-101	
EFP-1/3 Flow A OTSG	EF-25-FT	RECL-246	
EFP-1/3 Flow B OTSG	EF-23-FT	RECL-245	
EFP-2 Flow A OTSG	EF-26-FT	RECL-248	
EFP-2 Flow B OTSG	EF-24-FT	RECL-247	

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RB CONDITIONS

RB Sump Level (A Train)	WD-301A/B-LT	RECL-76	
RB Sump Level (B Train)	WD-302A/B-LT	RECL-75	
RB Flood Level (A Train)	WD-303A/B-LT	RECL-33	
RB Flood Level (B Train)	WD-304A/B-LT	RECL-34	

PARAMETER	REFERENCE/ SOURCE	REASON
RB High Rad Gamma Monitor	RM-G29	RECL-35
RB High Rad Gamma Monitor	RM-G30	RECL-36
RB Purge Exhaust Rad Monitor	RM-A1	RECL-155
RB Air Sample Rad Monitor	RM-A6	RECL-157
RB Hydrogen Concentration (A Train)	WS-011-CE	RECL-73
RB Hydrogen Concentration (B Train)	WS-010-CE	RECL-72
RB Temperature - 102 FT, 9 inches	AH-536-TE	RECL-77
RB Temperature - 125 FT	AH-537-TE	RECL-78
RB Temperature - 180 FT, 6 inches	AH-538-TE	RECL-80
RB Temperature - 235 FT	AH-539-TE	RECL-81
RB Average Temp		RECL-436
RB Pressure (A Loop)	BS-90-PT	RECL-82
RB Pressure (B Loop)	BS-91-PT	RECL-83
RB Pressure Narrow Range	BS-93-PT	RECL-84
Reactor Building Isolation	ESAS ACTUATION (SEE PAGE 8)	RECL-195 through 206 & RECL-368 through 373

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REACTIVITY

PARAMETER	REFERENCE/ SOURCE	REASON
Source Range NI-1, Computed		RECL-416
Source Range NI-2, Computed		RECL -417
Intermediate Range Detector NI-3		RECL-150
Intermediate Range Detector NI-4		RECL-151
NI Detector Power Range NI-5		RECL-0
NI Detector Power Range NI-6		RECL-1
NI Detector Power Range NI-7		RECL -2
NI Detector Power Range NI-8		RECL-3
Reactor Tripped		RECL-160
All Rods at In-Limits, computed		RECL-415

PUMP STATUS

PARAMETER	REFERENCE/ SOURCE	REASON
RCP Pump A Run Status	RCP-1A	RECL-162
RCP Pump B Run Status	RCP-1B	RECL-163
RCP Pump C Run Status	RCP-1C	RECL-164
RCP Pump D Run Status	RCP -1D	RECL-165
Decay Heat/LPI Pump DHP-1A		RECL-207

PARAMETER	REFERENCE/ SOURCE	REASON
Decay Heat/LPI Pump DHP-1B		RECL-208
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RADIATION		
RB High Rad Gamma Monitor	RM-G29	RECL-35
RB High Rad Gamma Monitor	RM-G30	RECL-36
RB Purge Exhaust Rad Monitor	RM-A1	RECL-155
RB Air Sample Rad Monitor	RM-A6	RECL-157
FH/Aux Bldg Exh. Rad Monitor	RM-A2	RECL-156
Primary Coolant Letdown Monitor	RM-L1	RECL-231
Steam Line A1 Rad Monitor	RM-G25	RECL-227
Steam Line B1 Rad Monitor	RM-G26	RECL-228
Steam Line A2 Rad Monitor	RM-G27	RECL-229
Steam Line B2 Rad Monitor	RM-G28	RECL-230
Condenser Vacuum Pump Exhaust	RM-A12	RECL-158
Plant Discharge Monitor	RM-L2	RECL-232
Turbine Building Sump Rad Monitor	RM-L7	RECL-233
Control Complex Rad Monitor	RM-A5	RECL-234
Control Room Monitor	RM-G1	RECL-235
PAGE 8		
ES TRIP STATUS		
RC1 ES CHAN A TRIPPED		RECL-195
RC2 ES CHAN A TRIPPED		RECL-196
RC3 ES CHAN A TRIPPED		RECL-197
RC4 ES CHAN A TRIPPED		RECL-198
RC5 ES CHAN A TRIPPED		RECL-199
RC6 ES CHAN A TRIPPED		RECL-200
RB1 ES CHAN A TRIPPED		RECL-201
RB2 ES CHAN A TRIPPED		RECL-202
RB3 ES CHAN A TRIPPED		RECL-203
RC1 ES CHAN B TRIPPED		RECL-204
RC2 ES CHAN B TRIPPED		RECL-205
RC3 ES CHAN B TRIPPED		RECL-206
RC4 ES CHAN B TRIPPED		RECL-368

PARAMETER	REFERENCE/ SOURCE	REASON
RC5 ES CHAN B TRIPPED RC6 ES CHAN B TRIPPED RB1 ES CHAN B TRIPPED RB2 ES CHAN B TRIPPED RB3 ES CHAN B TRIPPED		RECL-369 RECL-370 RECL-371 RECL-372 RECL-373
The SPDS need not be limited to the previously stated functions. It may include other functions that aid operating personnel in evaluating plant status.	53, 58	In addition to the three major types of information above to satisfy SPDS requirements, the system also incorporates a RECALL system which provides: A. A recording system to provide a record of the RECALL data one hour before an event and up to five hours following the event. B. Assurance that, upon occurrence of one of the "events" below, the signals above will be recorded and available for a period of at least 60 minutes prior to the event: 1. RC flow <55 Mlb/hr and reactor power >10%. 2. Total HPI flow > 750 GPM. 3. OTSG SU Level <18.75 inches. 4. Reactor Trip and rod group 1>50% withdrawn. 5. Pressurizer relief valve open. 6. Loss of 1 RC pump (pump status) and reactor power >10% 7. ICS FW cross limits and megawatts >20% 8. Reactor cross limits set. 9. Loss of both feedwater pumps and T-cold >400F. 10. Loss of 1 RC pump (RPM <1000) and reactor power >60% 11. Any two ES channels tripped. C. A data system that updates five times a second and is made available to the control room, TSC, and the EOF on a real time basis or from any previous time period.
The SPDS shall also be displayed in the TSC. The display(s) shall duplicate the	1, 59	The data available for display in the TSC must enable the plant

PARAMETER	REFERENCE/ SOURCE	REASON
display(s) of the control room.		management, engineering, and technical personnel assigned there to aid the control room operator in handling emergency conditions.
The data system shall provide access to accurate and reliable information sufficient to determine: • Plant Steady-state operating conditions prior to the accident • Transient conditions producing the initiating event, and • Plant systems dynamic behavior throughout the course of the accident	1, 2	The TSC data system may be used for: • Reviewing the accident sequence, • Determining appropriate mitigating actions, • Evaluation the extent of any damage, and • Determining plant status during recovery operations. NOTE: The NUREG 0696 "data system" in the TSC includes more than RECALL/SPDS. This design bases addresses only RECALL/SPDS part of the data system.
The SPDS display equipment used in the TSC need not be seismically qualified; it need only meet the TSC data system equipment reliability and performance criteria.	1	The SPDS is not a control system or a system with a safety-related function, but a reliable system for data recovery and display.
The design of the TSC data system equipment shall incorporate human factors engineering with consideration for both operating and maintenance personnel.	1, 59	This assures that call up, manipulation, and presentation of data can be easily performed and data display formats can be easily understood by the TSC personnel performing analyses.
Data storage and RECALL capability shall be provided for the TSC data set. At least 2 hours of post-event data shall be recorded. The sample frequency shall be chosen to be consistent with the use of the data. Capacity to record at least two weeks of additional post-event data with reduced time resolution shall be provided. Archival data storage and the capability to transfer data between active memory and archival data storage without interrupting TSC data acquisition and displays shall be provided for all TSC data.	1, 59, 64	To evaluate cause and effect of any event. Provide insight for corrective action. The data system provides a record of the RECALL data one hour before an event and up to five hours following the event.

PARAMETER	REFERENCE/ SOURCE	REASON
The SPDS shall also be displayed in the EOF. If the SPDS system in the control room is composed of multiple display units, multiple displays must also be provided in the EOF.	1, 59	This duplication will provide licensee management and NRC representatives information about the current reactor systems status and will facilitate communications among the control room, TSC, and EOF. Note: The SPDS displays are not duplicated in the EOF or required by this design guide. The intent of the requirement is met with the RECALL display system which makes all parameters used for SPDS display available in the EOF.
The design of the EOF data system equipment shall incorporate human-factors engineering with consideration for both operating and maintenance personnel.	1, 59	So that call up, manipulation, and presentation of data can be easily performed and data display formats can be easily understood by EOF personnel.
Data storage capability shall be provided for the EOF data set. At least 2 hours of pre-event data and 12 hours of post-event data shall be chosen to be consistent with the use of the data. Capacity to record at least two weeks of additional post-event data with reduced time resolution shall be provided. Archival data storage and the capability to transfer data between active memory and archival data storage without interrupting EOF data acquisition and displays shall be provided for all EOF data.	1, 59	Trend-information display and time-history display capability is required in the EOF to give EOF personnel a dynamic view of plant systems, radiological status, and environmental status during emergency.
Emergency Response Data System (ERDS) Provide real-time data transfer from plant computers to the NRC.	4a, 4b	The Emergency Response Data System permits the NRC to acquire data from CR3 in the event of an emergency.
ERDS Data Point Library (Attachment 1) Data Point Identifier, description, engineering units, range, alarms and/or technical specification limits and engineering system data.	4a	Site specific data base which provides the NRC Operations center information which relates the data both to the plant's design and to the manner in which the plant's Emergency Response Team utilizes and reacts to the data.
ERDS – Communication hardware (modem) capable of transmitting data at a minimum of 2400 BPS.	4a, 68	ERDS requires a standard modem capable of transferring data from 2400 BPS up to 14.4 BPS.

PARAMETER	REFERENCE/ SOURCE	REASON
Software – Operating System – Windows Application	68	Developed by Electronic Visions Inc. for Florida Power Corp.
Changes to software or hardware that affect the ERDS data points must be submitted to the NRC within 30 days after the changes are complete.	4a, 4b	To ensure system availability and operability.
Hardware or software changes, that could affect the transmission format and computer communications protocol to the ERDS must be provided to the NRC as soon as practicable and at least 30 days prior to the modification.	4a, 4b	To ensure system availability and operability.
The Licensee shall test the ERDS periodically to verify system availability and operability. The frequency of ERDS testing will be quarterly unless otherwise set by NRC based on demonstrated system performance.	4a, 4b	To ensure system availability and operability.

REFERENCES/SOURCE

Regulatory Documents
1) NUREG-0696, Functional Criteria for Emergency Response Facilities, dated February 1981
2) NUREG-0737, Supplement 1, Requirements for Engineering Response Capability (Generic Letter 82-33), Dated December 17, 1982.
3) Generic Letter 89-06
4) Regulatory Guide 1.97, "Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident"
4a) NUREG 1394, Emergency Response Data
4b) 10CFR50, Appendix E, Section VI, Emergency Response Data System

Modification Approval Records
5) MAR 81-6-38, "SPDS System Description"
6) MAR 86-04-20-01, "Reactor Trip High RC Pressure Trip Setpoint Increase."
7) MAR 91-07-13-02, "RECALL/SPDS Installation Upgrade"
8) MAR 96-07-17-01, "RCS Pressure Low Range Instrumentation Upgrade"
9) MAR 96-07-17-02, "Reg Guide 1.97 Low Range HPI Flow Range"
10) MAR 97-01-05-01, "Reg. Guide 1.97 LPI Flow Recording by Plant Computer"
10a) MAR 96-11-03-01, Subcooling Margin Monitor Upgrade

Calculations
11) F97-0013, "Pressure-Temperature Limits Report (PTLR), Rev. 1"
12) I84-0003, "SPDS Description Document"
13) I96-0002, "SPDS Tsat Display Error"
14) M97-0075, "CR3 NPSH/Seal Staging/DHRS/Subcooling/Surgeline/NDT Limits"
15) M97-0076, "CR3 P-T Curves Design Basis (OP-103B, Curves 4, 5, 6, and 7)"
16) M97-0141, "CR3 EOP Cooldown P-T Limits"

B&W (Framatome) Design Documents/Letters
17) 32-5000829-00, CR3 EOP Cooldown P-T Limits
18) 51-1121943-01 SPDS Safety Analysis, dated August 1984.
19) 51-1121942-11 SPDS Displays
20) 51-1152838-00, Technical Information for SPDS Inputs, dated December 4, 1984.
21) 74-1126473-00, Crystal River Unit 3 Abnormal Transient Operating Guidelines (ATOG) Part I and II, dated October 1, 1982.
22) 74-1123094, Operating Guidelines for Small Breaks for CR3, dated January 8, 1981.
23) 74-1152414-08, Emergency Operating Procedure Technical Basis Document, Vol. 3 Part II, Sect. 3.3, Page B-8.
24) Dwg# 1151659, FPC Safety Parameter Display System Functional Description, dated June 22, 1984
25) Dwg#1151640, FPC RECALL/SPDS Signal List
26) CR085-002, B&W Letter, SPDS Reliability Analysis, dated January 7, 1985.
27) FPC-88-745, B&W Letter, November 30 and December 1, 1989 Meeting Summary, dated 12/7/88
27a) FPC letter to USNRC, 3F1098-02, Subject: License Amendment Request #246, Revision 0, Subcooling Margin Monitoring Using Safety Parameter Display System, dated 10./30/98

FPC Letters/Correspondence
28) NEA-84-0193, April 16, 1984
29) NEA-84-0229, May 2, 1984

30) NEA-84-0270, May 25, 1984
31) NEA-84-0290, June 12, 1984
32) NEA-83-0333, July 19, 1984
33) **3F1083-27**, dated October 31, 1983, Subject: CR-3 NUREG-0737, Item I.D.1 Control Room Design Reviews
34) **3F0783-10**, dated July 12, 1983, Subject: CR-3 NUREG-0737, Supplement 1, Generic Letter 82-33 and Regulatory Guide 1.97, Rev. 3
35) **3F0884-08**, dated August 30, 1984, From: G.R. Westafer, manager Nuclear Operations and Fuel Management; To: John Stolz, Chief Operating Reactors Branch, Division of Licensing; Subject: NUREG-0737, Item I.D.2 and Supplement 1, Safety Parameter Display System
36) **3F0685-02**, dated June 17, 1985, Subject: NRC Confirmatory Order Dated February 21, 1984 NUREG-0737, Supplement 1.
37) **3F1085-12**, dated October 23, 1985, Subject: NUREG-0737, Item II.F.2, Inadequate Core Cooling Instrumentation Implementation Letter Report
38) **3F0686-23**, dated June 30, 1986, Subject: NUREG-0737, Supplement 1, Safety Parameter Display System.
39) **3F0886-02**, dated August, 1986, From: Rolf C. Widell, Manager Nuclear Operations Licensing and fuel Management; To: John Stolz, Director PWR Project Directorate #6 Division of PWR Licensing B, Office of Nuclear Reactor Regulation; Subject: NUREG-0737, Supplement 1, Safety Parameter Display System.
40) **3F0988-03**, dated September 1, 1988, From: Rolf c. Widell, Manager Nuclear Operations Licensing and Fuel Management; To: Document Control Desk, USNRC; Subject: Reactor Coolant Pump Trip, NUREG-0737, Item II.K.3.5, Request for Additional Information (TAC 49668)
41) **3F1288-13**, dated December 23, 1988, From: Rolf C. Widell, Director, Nuclear Operations Site Support; To: USNRC, Attn. Document Control Desk; Subject: Response to SER/TER Letter dated October 28, 1988, Safety Parameter Display System
42) **3F0789-06**, dated July 10, 1989, Subject: Safety Parameter Display System Generic Letter 89-06 (TAC No. 51233)
43) **3F1089-23**, dated October 31, 1989, Subject Technical Specification Change Request No. 174 - Pressure/Temperature Limits, Generic Letter 88-11 Submittal
44) **3F0190-06**, dated January 10, 1990, From: P.M. Beard, Jr. Senior Vice President, Nuclear Operations; To: USNRC, Attn. Document Control Desk; Subject: Confirmation of Conditions for High Pressure Injection (HPI) Flow Balancing
45) **3F0690-12**, dated June 21, 1990, From P.M. Beard, Jr. Senior Vice President, Nuclear Operations; To: USNRC, Attn. Document Control Desk; Subject: A) NRC to FPC letter dated October 28, 1988, "Crystal River Unit 3 (CR-3) Safety Parameter Display System (SPDS)," B) FPC to NRC letter 3F1288-13, dated December 23, 1988, C) NRC to FPC letter dated April 30, 1990, "Response to NRC Generic Letter 89-06 on the Safety Parameter Display System"
46) **3F0796-03**, dated July 8, 1996, Subject: Post-Accident Monitoring Instrumentation
47) **3F0996-05**, dated September 27, 1996, Subject: Technical Specification Change Request No. 209, Revision 0, Post Accident Monitoring Instrumentation
48) **IOC NED 95-0610**, dated October 26, 1995, "PICS Upgrade - SPDS Alerts"
49) **3F0797-21**, dated July 29, 1997, Subject, "Technical specification Change Request No. 209, Revision 1, Post-Accident Monitoring Instrumentation."

NRC Letters/Correspondence

50) NRC Letter dated September 6, 1985, Subject: Evaluation of the Detailed Control Room Design Review Summary Report
51) NRC Letter dated December 17, 1985, Subject: Request for Additional Information Concerning the CR-3 Safety Parameter Display System.

52) NRC Letter dated May 2, 1986, Subject: Request for Additional Information Concerning the CR-3 Safety Parameter Display System.
53) NRC Letter dated August 29, 1988, Subject: Detailed Control Room Design Review (DCRDR) CR-3 (TAC No. 56115).
54) NRC Letter dated September 10, 1987, Subject: TMI Action Item II.F.2 - Crystal River Unit 3 (TAC No. 45124)
55) **3N1088-15**, dated October 28, 1988, Subject: Crystal River Unit 3 (CR-3) Safety Parameter Display System (TAC No. 51233).
56) **3N0490-22**, dated April 30, 1990, Subject: Crystal River Unit 3 - Response to NRC Generic Letter 89-06 o the Safety Parameter Display System (TAC No's. 73649 and 51233)
57) **3N0491-09**, dated April 11, 1991, Subject: Notice of Violation (NRC Inspection Report No. 50-302/91-01)
58) **3N1297-17**, dated December 22, 1997, Subject, "Crystal River Unit 3 - Staff Evaluation and Issuance of Amendment RE: Post-Accident Monitoring Instrumentation (TAC No. M99308)"
58a) NRC Letter to FPC, Subject CR-3 – Staff Evaluation and Issuance of Amendment Regarding Subcooling Margin Monitoring Using SPDS (TAC No. MA4147), dated 4/20/99 (License Amendment No. 174).

FPC Procedures/Operating Manuals/Miscellaneous

59) RECALL/SPDS Operating Manual and Software Listings
60) SP-5095, April 29, 1983, Environmental and Seismic Qualification Guide
61) SP-5181, June 8, 1993, Plant Integrated Computer System
62) EP-220, Rev. 1, Pressurized Thermal Shock dated September 15, 1986
63) OP-103, Rev. 35, dated August 8, 1981, Plant Curve Book
64) OP-509, Rev. 8, Safety Parameter Display System Operation Procedure
65) FPC Purchase Order A-52294 and FPC Letter to D. Turner, Contract Variation, dated 3/23/81.
66) Replacement Operator Training Lesson, ROT-4-21, Plant Computer
67) FPC Drawing CR-3-164B, Sheets EM-1 through EM-17 and EM-24, Sheets 9 and 14 Rev. 0, Sheet 16 Rev. 2, all others Rev. 1.
68) Emergency Response Data System Interface, Vol. 1, Rev 0

Attachment 1

EMERGENCY REPONSE DATA SYSTEM POINTS LIST

	Point ID	Plant Specific Point Description
1.	P208	Linear Power Channel NI-5
2.	P209	Linear Power Channel NI-6
3.	P210	Linear Power Channel NI-7
4.	P211	Linear Power Channel NI-8
5.	P212	Log N Channel NI-3
6.	P213	Log N Channel NI-4
7.	P202	Log Count Rate Channel NI-1
8.	P203	Log Count Rate Channel NI-2
9.	R749	RC Loop A1 Flow
10.	R750	RC Loop A2 Flow
11.	R751	RC Loop B1 Flow
12.	R752	RC Loop B2 Flow
13.	S284	Steam Generator A Level (Operating)
14.	S336	Steam Generator B Level (Operating)
15.	S287	Steam Generator A Level (Start-up)
16.	S293	Steam Generator B Level (Start-up)
17.	S307	Steam Generator Outlet 1A Pressure
18.	S308	Steam Generator Outlet 1B Pressure
19.	S301	Steam Generator A Inlet Feedwater Flow
20.	S302	Steam Generator B Inlet Feedwater Flow
21.	S300	A Emergency Feedwater Flow
22.	S312	B Emergency Feedwater Flow
23.	R327	T-Hot Loop 164A (TY5)
24.	R328	T-Hot Loop 164B (TY5)
25.	R220	RC Pumps A Suction Temp (Wide)
26.	R221	RC Pumps B Suction Temp (Wide)
27.	R209	RC Press (Wide) (Loop A)
28.	R210	RC Press (Wide) (Loop B)
29.	R700	Pressurizer Level
30.	W324	Reactor Building Purge Duct – Low Range
31.	W325	Reactor Building Purge Duct – Mid Range
32.	W326	Reactor Building Purge Duct – High Range
33.	W329	Aux Bldg/Fuel Hand Bldg Purge Duct - Low
34.	W330	Aux Bldg/Fuel Hand Bldg Purge Duct - Mid
35.	W331	Aux Bldg/Fuel Hand Bldg Purge Duct - High
36.	W345	RM-L2 – Primary Plant Discharge Liquid
37.	W343	RM-A12 - Condenser Vacuum Pump Gas
38.	W322	RM-G29 – Inside Containment
39.	W323	RM-G30 – Inside Containment
40.	W344	RM-L1 – Primary Coolant Letdown
41.	W318	RM-G25 – A1 Main Steam Relief
42.	W319	RM-G26 – B1 Main Steam Relief
43.	W320	RM-G27 – A2 Main Steam Line
44.	W321	RM-G28 – B2 Main Steam Line
45.	P254	Reactor Building Pressure

46.	S837	Reactor Bldg Average Ambient Temperature
47.	W203	Wind Speed 33' Primary Tower
48.	W207	Wind Direction 33' Primary Tower

Attachment 1

EMERGENCY REPONSE DATA SYSTEM POINTS LIST (Continued)

49.	W421	Make-up Flow (GPM)
50.	W703	HPI Flow MUV-25 (GPM)
51.	W704	HPI Flow MUV-23 (GPM)
52.	W705	HPI Flow MUV-26 (GPM)
53.	W706	HPI Flow MUV-24 (GPM)
54.	W409	DH Flow A Pump (GPM)
55.	W410	DH Flow B Pump (GPM)
56.	W415	RB Sump B Level (Ft)
57.	W416	RB Sump A Level (Ft)
58.	W402	RB Sump A Fld Lvl
59.	W403	RB Sump B Fld Lvl
60.	W413	Hydrogen B (Percent)
61.	W414	Hydrogen A (Percent)
62.	X335	Borated Water Storage Tank Level
63.	R258	Incore Temperature 8-H
64.	R261	Incore Temperature 8-F
65.	R264	Incore Temperature 7-E
66.	R268	Incore Temperature 5-K
67.	R270	Incore Temperature 7-M
68.	R273	Incore Temperature 9-M
69.	R280	Incore Temperature 13-F
70.	R286	Incore Temperature 9-C
71.	R290	Incore Temperature 5-D
72.	R292	Incore Temperature 3-F
73.	R299	Incore Temperature 5-O
74.	R305	Incore Temperature12-O