REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:8302070273 DOC.DATE: 82/02/03 NOTARIZED: NO DOCKET #
FACIL:50-361 San Onofre Nuclear Station, Unit 2, Southern Californ 05000361
50-362 San Onofre Nuclear Station, Unit 3, Southern Californ 05000362
AUTH.NAME AUTHOR AFFILIATION
BASKIN,K.P. Southern California Edison Co.
RECIP.NAME RECIPIENT AFFILIATION
KNIGHTON,G.W. Licensing Branch 3

SUBJECT: Forwards responses to NRC 830114 request & clarified by 830123 meeting re ESF actuation sys event on 821217.Response to 14 addl questions,per NRC 830125 telcon also encl.

DISTRIBUTION CODE: B001S COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 2
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NOTES:J Hanchett 1cy PDR Documents. ELD Chandler 1cy. 05000361
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	NRR LB3 LA	1	0	ROOD,H. 01	1	1
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	IE/DEP EPDS 35	1	1	IE/DEP/EPLB 36	3	3
	NRR/DE/AEAB	1	0	NRR/DE/CEB 11	1	1
	NRR/DE/EQB 13	2	2	NRR/DE/GB 28	2	2
	NRR/DE/HGEB 30	1	1	NRR/DE/MEB 18	1	1
	NRR/DE/MTEB 17	1	1	NRR/DE/QAB 21	1	1
	NRR/DE/SAB 24	1	1	NRR/DE/SEB 25	1	1
	NRR/DHFS/HFEB40	1	1	NRR/DHFS/LQB 32	1	1
	NRR/DHFS/OLB 34	1	1	NRR/DL/SSPB	1	0
	NRR/DSI/AEB 26	1	1	NRR/DSI/ASB	1	1
	NRR/DSI/CPB 10	1	1	NRR/DSI/CSB 09	1	1
	NRR/DSI/ICSB 16	1	1	NRR/DSI/METB 12	1	1
	NRR/DSI/PSB 19	1	1	NRR/DSI/RAB 22	1	1
	NRR/DSI/RSB 23	1	1	REG FILE 04	1	1
	RGN5	3	3	RM/DDAMI/MIB	1	0
EXTERNAL:	ACRS 41	6	6	BNL(AMDTS ONLY)	1	1
	DMB/DSS (AMDTS)	1	1	FEMA-REP DIV 39	1	1
	LPDR 03	1	1	NRC PDR 02	1	1
	NSIC 05	1	1	NTIS	1	1
NOTES:		3	3			

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K. P. BASKIN
MANAGER OF NUCLEAR ENGINEERING,
SAFETY, AND LICENSING

TELEPHONE
(213) 572-1401

February 3, 1983

Director, Office of Nuclear Reactor Regulation
Attention: Mr. George W. Knighton, Branch Chief
Licensing Branch No. 3
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Subject: Docket Nos. 50-361 and 50-362
San Onofre Nuclear Generating Station
Units 2 and 3

SCE's letter of January 14, 1983 provided responses to the NRC's concerns relative to the December 17, 1982 Engineered Safety Features Actuation System (ESFAS) actuation event at San Onofre Unit 3. SCE subsequently met with the NRC on January 21, 1983 to discuss the responses provided in the January 14, 1983 letter.

During the January 21, 1983 meeting, SCE provided the staff with additional verbal clarification to support the January 14, 1983 submittal. The NRC requested SCE to formally document the verbal clarification provided during the meeting and in a subsequent conversation on January 25, 1983 the NRC staff requested SCE to provide responses to fourteen (14) additional questions.

In response to the NRC's requests, Enclosure I to this letter provides a report documenting the analyses and actions previously discussed in the January 14, 1983 letter amplified by the clarification provided during the January 21, 1983 meeting. Enclosure II responds to the fourteen additional questions provided by the NRC during the January 25, 1983 telephone conversation.

B001

SCE considers that the information provided by this submittal is sufficient to satisfy the NRC's concerns relative to SCE's resolution of this issue.

If you have any questions or comments, please let me know.

Very truly yours,

K P Baskin

Enclosures

cc: H. Rood(NRC)

ENCLOSURE I

ANALYSES AND ACTIONS RELATIVE TO THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS (ESFAS) ACTUATION EVENT AT SAN ONOFRE UNIT 3 ON DECEMBER 17, 1982

February 1983

ANALYSES AND ACTIONS RELATIVE TO THE ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS (ESFAS) ACTUATION EVENT AT SAN ONOFRE UNIT 3 ON DECEMBER 17, 1982

OUTLINE

Section 1 - PPS Instrumentation Review

1.1 Hardware Changes

As a result of the December 17, 1982 spurious ESFAS initiation event on San Onofre Unit 3, SCE undertook two tasks to investigate the PPS performance.

One task force headed by CE engineers performed a thorough design review of the PPS components and interfacing connection to look for a common point where a fault could potentially result in a similar event scenario. The only problem identified was in the wiring of the J3109 connectors in PPS channels A and D. Momentary disconnection of either of these connectors would cause a complete ESFAS actuation. A discussion of the task force's conclusion was presented in Enclosure 1 to SCE's letter of January 14, 1983.

The other task force headed by SCE station engineers performed a thorough investigation of the PPS hardware and reconstructed the event scenario as accurately as possible in an attempt to identify potential causes. The task force's conclusion was that two independent failures on different PPS power supplies caused the inadvertant ESFAS actuation.

The actions taken based on the task force results were as follows:

(a) A design change package (DCP) has been prepared to accomplish the J3109 connector modification to insure that upon momentary connector disconnection an ESFAS actuation will not occur. In response to the NRC's request for clarification during the January 25, 1983 telephone conversation, copies of the DCPs for San Onofre Unit 2 and Unit 3 are included as Section 1.2 of this report.

(b) The PPS components identified by the station task force as potentially causing the independent failures have been repaired. This consisted of replacing a switch which provided channel A vital bus power to the matrix logic and tightening the loose leads on a Channel D PPS power supply which supplies the same matrix logic.

The DCPs included as Section 1.2 of this report provide the details for the connector wiring change. The test procedures associated with the DCP for the connector modification include normal 31 day matrix relay surveillance testing and disconnecting of the J3109 connectors to verify that a full trip does not occur.

SECTION 1.2

DESIGN CHANGE PACKAGES FOR
PPS CONNECTOR MODIFICATIONS

DESIGN CHANGE EVALUATION REPORT

HEADING

DCP NUMBER 1470-DCP-123 ISSUE DATE 1/3/83 DEPT/SECTION/GROUP I&CE

UNIT AFFECTED 3 FAR REFERENCE 1470-526 COMPONENT/SYSTEM Plant Protection System (PPS)

USE SPACE PROVIDED OR ATTACH SEPARATE SHEETS AS NECESSARY.

DESCRIPTION OF CHANGE

See Attachment A.

REASON/PURPOSE FOR CHANGE

See Attachment A.

ACTIONS REQUIRED

See Attached Wiring Change Procedure.

DESIGN CRITERIA INVOLVED (CHECK AS APPLICABLE; LIST IN REFERENCES)

( )GOVERNING REGULATIONS ( )ENVIRONMENT/ACCIDENT CRITERIA ( )OTHERS
(XX)INPUT DESIGN PARAMETERS ( )PLANT OPERATIONS CRITERIA

EVALUATION TO 10CFR50.59 THE DESIGN CHANGE RELATIVE TO 10CFR50.59:

A. (DOES) DOES NOT INCREASE THE PROBABILITY OF OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT. REASON:

See Attachment B, page 4.

B. (DOES) DOES NOT CREATE A POSSIBILITY FOR AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN ANY EVALUATED PREVIOUSLY IN THE SAFETY ANALYSIS REPORT. REASON:

See A above.

C. (DOES) DOES NOT REDUCE THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATIONS. REASON:

See A above.

DOCUMENTS AFFECTED (CHECK AS APPLICABLE AND LIST ON ATTACHMENT)

( )COMPONENT SPECIFICATIONS ( )SAR MATERIAL ( )PRIOR DCP'S
( )TEST REQUIREMENTS/GUIDELINES ( )OPERATING PROCEDURES ( )TECH. MANUALS
( )TECHNICAL SPECIFICATIONS (XX)DRAWINGS/SCHEMATICS ( )OTHER

INTERFACES THE DESIGN CHANGE PACKAGE:

A. (DOES) DOES NOT INVOLVE INTERFACES WITH C-E (ANOTHER GROUP).
B. (DOES) DOES NOT INVOLVE A BOP INTERFACE. (DESCRIPTION OF IMPACT IS ATTACHED.)
C. IS (IS NOT) A QUALITY RECORD.

TECHNICAL GROUP APPROVALS DCP REPORT MATERIAL IS COMPLETE WITH ALL NECESSARY ATTACHMENTS.
(NAMES TYPED AND SIGNED)

D.R. Waite D R Waite	1-6-83	P.L. Yanosy P L Yanosy	1/6/83
PREPARED, COGNIZANT ENGINEER, LEAD GROUP	DATE	APPROVAL, SUPERVISOR, LEAD GROUP	DATE
N/A		N/A	
APPROVAL, SUPERVISOR, SUPPORT GROUP	DATE	APPROVAL, SUPERVISOR, SUPPORT GROUP	DATE

PART I

DESIGN CHANGE PACKAGE DOCUMENTATION

DCP NUMBER 1470-DCP-123

ITEM NO.	DOCUMENT DESCRIPTION (NUMBER & TITLE)	REV.	REMARKS / ACTIONS
1	PPS Technical Manual	B	
	Figure 8-19		Fig. 8-19 is Rev.00
	Figure 8-20		Fig. 8-20 is Rev.00
	Figure 8-21		Fig. 8-21 is Rev.00
2	PPS Relay Card Rack Channel A		
	Wire List 30370 pp. 40-43	10	
3	PPS Relay Card Rack Channel D		
	Wire List 30373 pp. 40-43	11	

REMARKS / ACTIONS:

Items 1-3 reflect pre-DCP status of the affected areas and are not required for implementation of this DCP. However, due to the large number of reterminations involved, items 1-3 will be revised and transmitted at a later date.

LIST OF REFERENCES (NOT INCLUDED IN PACKAGE)

1) CE FAR 1470- 526

ATTACHMENT A

DESCRIPTION/PURPOSE OF CHANGE

This DCP documents wiring modifications to be made in the PPS Relay Card Rack connectors AJ3109 and DJ3109. THe present as-wired condition of connectors AJ3109 and DJ3109 is such that the single disconnection of either connector will result in full Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) initiations. The original C-E design intent was such that the disconnection of a single connector would not cause a system actuation. Therefore, these two connectors violate this intent. The wiring changes to be made are such that the single disconnection of either connector will not cause full RPS and ESFAS initiations. A design review of the PPS has indicated that no other connector when singly disconnected will cause full RPS and/or ESFAS initiations; only Channels A & D are affected.

WIRING CHANGE PROCEDURE

Perform the wiring changes indicated on pages 5 and 6. Perform a subsequent continuity check to verify the correct wiring. Also, after completion of these modifications, perform the following test:

1) Disconnect the AP3109 connector from AJ3109. Verify that full RPS and ESFAS initations have not occurred. This is done by observing that the Status Panel lights for the RPS and ESFAS are still illuminated. Reconnect the AP3109 connector to AJ3109.

2) Repeat Step 1 above for the DP3109/DJ3109 connector.

The wiring modifications listed in this procedure swap wires between the connectors AJ3109 and AJ3110 and also between the connectors DJ3109 and DJ3110. The wiring changes prevent all four RPS and ESFAS trip paths from being broken upon removal of either single connector, thus eliminating full RPS and ESFAS initiations.

NOTE: A modification to prevent the automatic closure of Safety Injection System (SIS) mini-flow valves on a Recirculation Actuation Signal (RAS) is separately addressed in 1470-DCP-122.

EVALUATION TO 10CFR50.59

This design change prevents a single PPS connector disconnection from causing full PPS and ESFAS initiations,as intended by the original system design. As a result:

1. The probability of occurrence or consequences of an accident or malfunction previously evaluated in the FSAR is not increased. ESFAS actuation causes sequences of valve and component operation to occur which place the plant in an operating configuration suitable for mitigating the consequences of the design basis accidents analyzed in the FSAR. Open curcuit at the connectors as they currently exist does not in itself create a situation outside the bounds of the existing safety analysis. Modification of connectors AJ3109,AJ3110, DJ3109, and DJ3110 by rerouting of circuitry will not create the possibility of any new or unanalyzed accidents or conditions and will improve the overall reliability of the safety systems by decreasing the possibility of spurious challenges. The combination of circuits through the four connectors (resulting from the modification) has been reviewed. It has been determined that no unacceptable conditions will result if any of the four modified connectors suffer loss of continuity. Thus, rerouting of the actuation curcuitry will not increase the consequences of an accident. As PPS inputs and actuation logics are not being modified as a result of this change, no change will result in the overall function of the PPS. Post implementation testing will provide assurance of this. It is concluded that the probability of spurious challenges to the safety systems presented by inadvertant ESFAS actuation is reduced.

2. The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR is not created. The circuit reconfiguration will bring the PPS into conformance with its original system design intent and philosophy. Thus the system will be physically configured and will perform as assumed in the FSAR safety analysis. Installation verification testing will demonstrate that continuity interruption at one of the modified connectors (AJ3109, AJ3110, DJ3109, and DJ3110) will not cause ESFAS actuation. PPS operability will be verified by performance of established surveillance testing before the channels are returned to service.

3. The margin of safety as defined in the basis of the Technical Specifications is not reduced. However, the overall margin of safety provided to the public is increased. Prevention of spurious challenges to the safety systems will increase the overall reliability of those systems thus providing a higher degree of assurance they will be available to perform their intended safety functions if required.

977-137-0

1470-DCP-123

Pre DCP Wiring				Post DCP Wiring	
	From Termination	To Termination		From Termination	To Termination
1)	AJ3109-49	AJ3075-02	must be changed to	AJ3110-29	AJ3075-02
	AJ3110-29	AJ3111W	must be changed to	AJ3109-49	AJ3111W
2)	AJ3109-50	AJ3076-02	must be changed to	AJ3110-30	AJ3076-02
	AJ3110-30	AJ3074-02	must be changed to	AJ3109-50	AJ3074-02
3)	AJ3109-51	AJ3079-02	must be changed to	AJ3110-33	AJ3079-02
	AJ3110-33	AJ3111-K	must be changed to	AJ3109-51	AJ3111-K
4)	AJ3109-52	AJ3080-02	must be changed to	AJ3110-34	AJ3080-02
	AJ3110-34	AJ3078-02	must be changed to	AJ3109-52	AJ3078-02
5)	AJ3109-53	AJ3079-13	must be changed to	AJ3110-37	AJ3079-13
	AJ3110-37	AJ3017-05	must be changed to	AJ3109-53	AJ3017-05
6)	AJ3109-54	AJ3080-13	must be changed to	AJ3110-38	AJ3080-13
	AJ3110-38	AJ3078-13	must be changed to	AJ3109-54	AJ3078-13
7)	AJ3109-55	AJ3079-20	must be changed to	AJ3110-41	AJ3079-20
	AJ3110-41	AJ3017-34	must be changed to	AJ3109-55	AJ3017-34
8)	AJ3109-56	AJ3080-20	must be changed to	AJ3110-42	AJ3080-20
	AJ3110-42	AJ3078-20	must be changed to	AJ3109-56	AJ3078-20
9)	AJ3109-57	AJ3079-31	must be changed to	AJ3110-45	AJ3079-31
	AJ3110-45	AJ3111-C	must be changed to	AJ3109-57	AJ3111-C
10)	AJ3109-58	AJ3080-31	must be changed to	AJ3110-46	AJ3080-31
	AJ3110-46	AJ3078-31	must be changed to	AJ3109-58	AJ3078-31
11)	AJ3109-59	AJ3083-02	must be changed to	AJ3110-49	AJ3083-02
	AJ3110-49	AJ3111-E	must be changed to	AJ3109-59	AJ3111-E
12)	AJ3109-60	AJ3084-02	must be changed to	AJ3110-50	AJ3084-02
	AJ3110-50	AJ3082-02	must be changed to	AJ3109-60	AJ3082-02
13)	AJ3109-61	AJ3083-13	must be changed to	AJ3110-53	AJ3083-13
	AJ3110-53	AJ3111-M	must be changed to	AJ3109-61	AJ3111-M
14)	AJ3109-62	AJ3084-13	must be changed to	AJ3110-54	AJ3084-13
	AJ3110-54	AJ3082-13	must be changed to	AJ3109-62	AJ3082-13
15)	AJ3109-63	AJ3083-20	must be changed to	AJ3110-57	AJ3083-20
	AJ3110-57	AJ3111-P	must be changed to	AJ3109-53	AJ3111-P
16)	AJ3109-64	AJ3084-20	must be changed to	AJ3110-58	AJ3084-20
	AJ3110-58	AJ3082-20	must be changed to	AJ3109-64	AJ3082-20

977-137-0

Pre DCP Wiring			Post DCP Wiring	
From Termination	To Termination		From Termination	To Termination
1) DJ3109-49	DJ3075-34	must be changed to	DJ3110-29	DJ3075-34
DJ3110-29	DJ3073-34	must be changed to	DJ3109-49	DJ3073-34
2) DJ3109-50	DJ3111-W	must be changed to	DJ3110-30	DJ3111W
DJ3110-30	DJ3074-34	must be changed to	DJ3109-50	DJ3074-34
3) DJ3109-51	DJ3079-34	must be changed to	DJ3110-33	DJ3079-34
DJ3110-33	DJ3077-34	must be changed to	DJ3109-51	DJ3077-34
4) DJ3109-52	DJ3111-K	must be changed to	DJ3110-34	DJ3111-K
DJ3110-34	DJ3078-34	must be changed to	DJ3109-52	DJ3078-34
5) DJ3109-53	DJ3079-47	must be changed to	DJ3110-37	DJ3079-47
DJ3110-37	DJ3077-47	must be changed to	DJ3109-53	DJ3077-47
6) DJ3109-54	DJ3017-05	must be changed to	DJ3110-38	DJ3017-05
DJ3110-38	DJ3078-47	must be changed to	DJ3109-54	DJ3078-47
7) DJ3109-55	DJ3079-50	must be changed to	DJ3110-41	DJ3079-50
DJ3110-41	DJ3077-50	must be changed to	DJ3109-55	DJ3077-50
8) DJ3109-56	DJ3017-02	must be changed to	DJ3110-42	DJ3017-02
DJ3110-42	DJ3078-50	must be changed to	DJ3109-56	DJ3078-50
9) DJ3109-57	DJ3079-63	must be changed to	DJ3110-45	DJ3079-63
DJ3110-45	DJ3077-63	must be changed to	DJ3109-57	DJ3077-63
10) DJ3109-58	DJ3111-C	must be changed to	DJ3110-46	DJ3111-C
DJ3110-46	DJ3078-63	must be changed to	DJ3109-58	DJ3078-63
11) DJ3109-59	DJ3083-34	must be changed to	DJ3110-49	DJ3083-34
DJ3110-49	DJ3081-34	must be changed to	DJ3109-59	DJ3081-34
12) DJ3109-60	DJ3111-E	must be changed to	DJ3110-50	DJ3111-E
DJ3110-50	DJ3082-34	must be changed to	DJ3109-60	DJ3082-34
13) DJ3109-61	DJ3083-47	must be changed to	DJ3110-53	DJ3083-47
DJ3110-53	DJ3081-47	must be changed to	DJ3109-61	DJ3081-47
14) DJ3109-62	DJ3111-M	must be changed to	DJ3110-54	DJ3111-M
DJ3110-54	DJ3082-47	must be changed to	DJ3109-62	DJ3082-47
15) DJ3109-63	DJ3083-50	must be changed to	DJ3110-57	DJ3083-50
DJ3110-57	DJ3081-50	must be changed to	DJ3109-63	DJ3081-50
16) DJ3109-64	DJ3111-P	must be changed to	DJ3110-58	DJ3111-P
DJ3110-58	DJ3082-50	must be changed to	DJ3109-64	DJ3082-50

REVIEW COMMITTEE APPROVAL

DCP NUMBER 1470-DCP-123

1. DEPARTMENT/SECTION/GROUP APPROVAL(S) COMPLETE.
2. REVIEW COMMITTEE HAS REVIEWED THE DESIGN CHANGE PACKAGE FOR COMPLETENESS AND CONSISTENCY WITH PREVIOUS EVALUATIONS. TECHNICAL POSITIONS TO 10CFR50.59 ARE ADEQUATELY PRESENTED.
3. THERE IS CONCURRENCE WITH THE C-E SITE OFFICE ON THE DETAILS OF IMPLEMENTATION.
4. APPROVALS:

Robert E Keen APPLICATION ENGINEER	1-6-83 DATE	ICE DEPT/SECTION/GROUP
JLBarry APPLICATION ENGINEER	1-7-83 DATE	MD DEPT/SECTION/GROUP
W. AB Johnson APPLICATION ENGINEER	1-10-83 DATE	RD DEPT/SECTION/GROUP
[signature] APPLICATION ENGINEER	1-10-83 DATE	PE DEPT/SECTION/GROUP
V Dmulli SYSTEMS ENGINEERING REPRESENTATIVE	1/11/83 DATE	
[signature] NUCLEAR LICENSING REPRESENTATIVE	7 JAN 83 DATE	
Fred Bennett PROJECT OFFICE REPRESENTATIVE	1/13/83 DATE	
OTHERS (AS SPECIFIED) P.C. Rohr	1-7-83 DATE	NSSS Test
	DATE	

IMPLEMENTATION

5. FIELD IMPLEMENTATION OF DESIGN CHANGE (INFORMATION VIA PROJECT OFFICE/SITE)
   - A. WORK TO BE PERFORMED BY ____________
   - B. CONSTRUCTION SAFETY EVALUATION BY ____________
   - C. MATERIALS/COMPONENTS REQUIRED ( )YES ( )NO IF YES, EXPLAIN.
   - D. SPARE PARTS AFFECTED. ( )YES ( )NO IF YES, EXPLAIN.
   - E. PO/MO NUMBER, IF APPLICABLE ____________

   FIELD IMPLEMENTATION COMPLETED.

   ____________ PROJECT OFFICE ____________ DATE

6. ENGINEERING IMPLEMENTATION COMPLETED.

   ____________ SYSTEMS ENGINEERING ____________ DATE

DESIGN CHANGE EVALUATION REPORT

DING

CP NUMBER 1370-DCP-130 ISSUE DATE 1/3/83 DEPT/SECTION/GROUP I&CE

SCE UNIT AFFECTED 2 FAR REFERENCE 1370-829 COMPONENT/SYSTEM Plant Protection System (PPS)

USE SPACE PROVIDED OR ATTACH SEPARATE SHEETS AS NECESSARY.

DESCRIPTION OF CHANGE

See Attachment A.

REASON/PURPOSE FOR CHANGE

See Attachment A.

ACTIONS REQUIRED

See Attached Wiring Change Procedure.

DESIGN CRITERIA INVOLVED (CHECK AS APPLICABLE; LIST IN REFERENCES)

( )GOVERNING REGULATIONS ( )ENVIRONMENT/ACCIDENT CRITERIA ( )OTHERS

(XX)INPUT DESIGN PARAMETERS ( )PLANT OPERATIONS CRITERIA

EVALUATION TO 10CFR50.59 THE DESIGN CHANGE RELATIVE TO 10CFR50.59:

A. (DOES) DOES NOT INCREASE THE PROBABILITY OF OCCURRENCE OR THE CONSEQUENCES OF AN ACCIDENT OR MALFUNCTION OF EQUIPMENT IMPORTANT TO SAFETY PREVIOUSLY EVALUATED IN THE SAFETY ANALYSIS REPORT. REASON:

See Attachment B, page 4.

(DOES) DOES NOT CREATE A POSSIBILITY FOR AN ACCIDENT OR MALFUNCTION OF A DIFFERENT TYPE THAN ANY EVALUATED PREVIOUSLY IN THE SAFETY ANALYSIS REPORT. REASON:

See A above.

C. (DOES) DOES NOT REDUCE THE MARGIN OF SAFETY AS DEFINED IN THE BASIS FOR ANY TECHNICAL SPECIFICATIONS. REASON:

See A above.

DOCUMENTS AFFECTED (CHECK AS APPLICABLE AND LIST ON ATTACHMENT)

( )COMPONENT SPECIFICATIONS ( )SAR MATERIAL ( )PRIOR DCP'S

( )TEST REQUIREMENTS/GUIDELINES ( )OPERATING PROCEDURES ( )TECH. MANUALS

( )TECHNICAL SPECIFICATIONS (XX)DRAWINGS/SCHEMATICS ( )OTHER

INTERFACES THE DESIGN CHANGE PACKAGE:

A. (DOES) DOES NOT INVOLVE INTERFACES WITH C-E (ANOTHER GROUP).

B. (DOES) DOES NOT INVOLVE A BOP INTERFACE. (DESCRIPTION OF IMPACT IS ATTACHED.)

C. IS (IS NOT) A QUALITY RECORD.

TECHNICAL GROUP APPROVALS DCP REPORT MATERIAL IS COMPLETE WITH ALL NECESSARY ATTACHMENTS.
(NAMES TYPED AND SIGNED)

D.R. Waite D R Waite 1-6-83 P.L. Yanosy P L Yanosy 1/6/83

PREPARED, COGNIZANT ENGINEER, LEAD GROUP DATE APPROVAL, SUPERVISOR, LEAD GROUP DATE

N/A N/A

PPROVAL, SUPERVISOR, SUPPORT GROUP DATE APPROVAL, SUPERVISOR, SUPPORT GROUP DATE

PART I

DESIGN CHANGE PACKAGE DOCUMENTATION

DCP NUMBER 1370-DCP-130

ITEM NO.	DOCUMENT DESCRIPTION (NUMBER & TITLE)	REV.	REMARKS / ACTIONS
1	PPS Technical Manual	B	
	Figure 8-19		Fig. 8-19 is Rev.00
	Figure 8-20		Fig. 8-20 is Rev.00
	Figure 8-21		Fig. 8-21 is Rev.00
2	PPS Relay Card Rack Channel A		
	Wire List 30370 Rev. 10 pp 40-43	10	
3	PPS Relay Card Rack Channel D		
	Wire List 30373 Rev. 11 pp 40-43	11	

REMARKS / ACTIONS:

Items 1-3 which reflect pre-DCP status of the affected areas, are not required for implementation of this DCP. However, due to the large number of reterminations involved, items 1-3 will be revised and transmitted at a later date.

LIST OF REFERENCES (NOT INCLUDED IN PACKAGE)

1) CE FAR 1370-829

ATTACHMENT A

DESCRIPTION/PURPOSE OF CHANGE

This DCP documents wiring modifications to be made in the PPS Relay Card Rack connectors AJ3109 and DJ3109. THe present as-wired condition of connectors AJ3109 and DJ3109 is such that the single disconnection of either connector will result in full Reactor Protection System (RPS) and Engineered Safety Features Actuation System (ESFAS) initiations. The original C-E design intent was such that the disconnection of a single connector would not cause a system actuation. Therefore, these two connectors violate this intent. The wiring changes to be made are such that the single disconnection of either connector will not cause full RPS and ESFAS initiations. A design review of the PPS has indicated that no other connector when singly disconnected will cause full RPS and/or ESFAS initiations; only Channels A & D are affected.

WIRING CHANGE PROCEDURE

Perform the wiring changes indicated on pages 5 and 6. Perform a subsequent continuity check to verify the correct wiring. Also, after completion of these modifications, perform the following test:

1) Disconnect the AP3109 connector from AJ3109. Verify that full RPS and ESFAS initations have not occurred. This is done by observing that the Status Panel lights for the RPS and ESFAS are still illuminated. Reconnect the AP3109 connector to AJ3109.

2) Repeat Step 1 above for the DP3109/DJ3109 connector.

The wiring modifications listed in this procedure swap wires between the connectors AJ3109 and AJ3110 and also between the connectors DJ3109 and DJ3110. The wiring changes prevent all four RPS and ESFAS trip paths from being broken upon removal of either single connector, thus eliminating full RPS and ESFAS initiations.

NOTE: A modification to prevent the automatic closure of Safety Injection System (SIS) mini-flow valves on a Recirculation Actuation Signal (RAS) is separately addressed in 1370-DCP-129

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Attachment B

EVALUATION TO 10CFR50.59

This design change prevents a single PPS connector disconnection from causing full PPS and ESFAS initiations,as intended by the original system design. As a result:

1. The probability of occurrence or consequences of an accident or malfunction previously evaluated in the FSAR is not increased. ESFAS actuation causes sequences of valve and component operation to occur which place the plant in an operating configuration suitable for mitigating the consequences of the design basis accidents analyzed in the FSAR. Open curcuit at the connectors as they currently exist does not in itself create a situation outside the bounds of the existing safety analysis. Modification of connectors AJ3109,AJ3110, DJ3109, and DJ3110 by rerouting of circuitry will not create the possibility of any new or unanalyzed accidents or conditions and will improve the overall reliability of the safety systems by decreasing the possibility of spurious challenges. The combination of circuits through the four connectors (resulting from the modification) has been reviewed. It has been determined that no unacceptable conditions will result if any of the four modified connectors suffer loss of continuity. Thus, rerouting of the actuation curcuitry will not increase the consequences of an accident. As PPS inputs and actuation logics are not being modified as a result of this change, no change will result in the overall function of the PPS. Post implementation testing will provide assurance of this. It is concluded that the probability of spurious challenges to the safety systems presented by inadvertant ESFAS actuation is reduced.

2. The possibility for an accident or malfunction of a different type than previously evaluated in the FSAR is not created. The circuit reconfiguration will bring the PPS into conformance with its original system design intent and philosophy. Thus the system will be physically configured and will perform as assumed in the FSAR safety analysis. Installation verification testing will demonstrate that continuity interruption at one of the modified connectors (AJ3109, AJ3110, DJ3109, and DJ3110) will not cause ESFAS actuation. PPS operability will be verified by performance of established surveillance testing before the channels are returned to service.

3. The margin of safety as defined in the basis of the Technical Specifications is not reduced. However, the overall margin of safety provided to the public is increased. Prevention of spurious challenges to the safety systems will increase the overall reliability of those systems thus providing a higher degree of assurance they will be available to perform their intended safety functions if required.

CHANNEL A WIRING CHANGES

	Pre DCP Wiring: From Termination	Pre DCP Wiring: To Termination		Post DCP Wiring: From Termination	Post DCP Wiring: To Termination
1)	AJ3109-49	AJ3075-02	must be changed to	AJ3110-29	AJ3075-02
	AJ3110-29	AJ3111W	must be changed to	AJ3109-49	AJ3111W
2)	AJ3109-50	AJ3076-02	must be changed to	AJ3110-30	AJ3076-02
	AJ3110-30	AJ3074-02	must be changed to	AJ3109-50	AJ3074-02
3)	AJ3109-51	AJ3079-02	must be changed to	AJ3110-33	AJ3079-02
	AJ3110-33	AJ3111-K	must be changed to	AJ3109-51	AJ3111-K
4)	AJ3109-52	AJ3080-02	must be changed to	AJ3110-34	AJ3080-02
	AJ3110-34	AJ3078-02	must be changed to	AJ3109-52	AJ3078-02
5)	AJ3109-53	AJ3079-13	must be changed to	AJ3110-37	AJ3079-13
	AJ3110-37	AJ3017-05	must be changed to	AJ3109-53	AJ3017-05
6)	AJ3109-54	AJ3080-13	must be changed to	AJ3110-38	AJ3080-13
	AJ3110-38	AJ3078-13	must be changed to	AJ3109-54	AJ3078-13
7)	AJ3109-55	AJ3079-20	must be changed to	AJ3110-41	AJ3079-20
	AJ3110-41	AJ3017-34	must be changed to	AJ3109-55	AJ3017-34
8)	AJ3109-56	AJ3080-20	must be changed to	AJ3110-42	AJ3080-20
	AJ3110-42	AJ3078-20	must be changed to	AJ3109-56	AJ3078-20
9)	AJ3109-57	AJ3079-31	must be changed to	AJ3110-45	AJ3079-31
	AJ3110-45	AJ3111-C	must be changed to	AJ3109-57	AJ3111-C
10)	AJ3109-58	AJ3080-31	must be changed to	AJ3110-46	AJ3080-31
	AJ3110-46	AJ3078-31	must be changed to	AJ3109-58	AJ3078-31
11)	AJ3109-59	AJ3083-02	must be changed to	AJ3110-49	AJ3083-02
	AJ3110-49	AJ3111-E	must be changed to	AJ3109-59	AJ3111-E
12)	AJ3109-60	AJ3084-02	must be changed to	AJ3110-50	AJ3084-02
	AJ3110-50	AJ3082-02	must be changed to	AJ3109-60	AJ3082-02
13)	AJ3109-61	AJ3083-13	must be changed to	AJ3110-53	AJ3083-13
	AJ3110-53	AJ3111-M	must be changed to	AJ3109-61	AJ3111-M
14)	AJ3109-62	AJ3084-13	must be changed to	AJ3110-54	AJ3084-13
	AJ3110-54	AJ3082-13	must be changed to	AJ3109-62	AJ3082-13
15)	AJ3109-63	AJ3083-20	must be changed to	AJ3110-57	AJ3083-20
	AJ3110-57	AJ3111-P	must be changed to	AJ3109-53	AJ3111-P
16)	AJ3109-64	AJ3084-20	must be changed to	AJ3110-58	AJ3084-20
	AJ3110-58	AJ3082-20	must be changed to	AJ3109-64	AJ3082-20

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CHANNEL D WIRING CHANGES

	Pre DCP Wiring From Termination	To Termination		Post DCP Wiring From Termination	To Termination
1)	DJ3109-49	DJ3075-34	must be changed to	DJ3110-29	DJ3075-34
	DJ3110-29	DJ3073-34	must be changed to	DJ3109-49	DJ3073-34
2)	DJ3109-50	DJ3111-W	must be changed to	DJ3110-30	DJ3111W
	DJ3110-30	DJ3074-34	must be changed to	DJ3109-50	DJ3074-34
3)	DJ3109-51	DJ3079-34	must be changed to	DJ3110-33	DJ3079-34
	DJ3110-33	DJ3077-34	must be changed to	DJ3109-51	DJ3077-34
4)	DJ3109-52	DJ3111-K	must be changed to	DJ3110-34	DJ3111-K
	DJ3110-34	DJ3078-34	must be changed to	DJ3109-52	DJ3078-34
5)	DJ3109-53	DJ3079-47	must be changed to	DJ3110-37	DJ3079-47
	DJ3110-37	DJ3077-47	must be changed to	DJ3109-53	DJ3077-47
6)	DJ3109-54	DJ3017-05	must be changed to	DJ3110-38	DJ3017-05
	DJ3110-38	DJ3078-47	must be changed to	DJ3109-54	DJ3078-47
7)	DJ3109-55	DJ3079-50	must be changed to	DJ3110-41	DJ3079-50
	DJ3110-41	DJ3077-50	must be changed to	DJ3109-55	DJ3077-50
8)	DJ3109-56	DJ3017-02	must be changed to	DJ3110-42	DJ3017-02
	DJ3110-42	DJ3078-50	must be changed to	DJ3109-56	DJ3078-50
9)	DJ3109-57	DJ3079-63	must be changed to	DJ3110-45	DJ3079-63
	DJ3110-45	DJ3077-63	must be changed to	DJ3109-57	DJ3077-63
10)	DJ3109-58	DJ3111-C	must be changed to	DJ3110-46	DJ3111-C
	DJ3110-46	DJ3078-63	must be changed to	DJ3109-58	DJ3078-63
11)	DJ3109-59	DJ3083-34	must be changed to	DJ3110-49	DJ3083-34
	DJ3110-49	DJ3081-34	must be changed to	DJ3109-59	DJ3081-34
12)	DJ3109-60	DJ3111-E	must be changed to	DJ3110-50	DJ3111-E
	DJ3110-50	DJ3082-34	must be changed to	DJ3109-60	DJ3082-34
13)	DJ3109-61	DJ3083-47	must be changed to	DJ3110-53	DJ3083-47
	DJ3110-53	DJ3081-47	must be changed to	DJ3109-61	DJ3081-47
14)	DJ3109-62	DJ3111-M	must be changed to	DJ3110-54	DJ3111-M
	DJ3110-54	DJ3082-47	must be changed to	DJ3109-62	DJ3082-47
15)	DJ3109-63	DJ3083-50	must be changed to	DJ3110-57	DJ3083-50
	DJ3110-57	DJ3081-50	must be changed to	DJ3109-63	DJ3081-50
16)	DJ3109-64	DJ3111-P	must be changed to	DJ3110-58	DJ3111-P
	DJ3110-58	DJ3082-50	must be changed to	DJ3109-64	DJ3082-50

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Review Committee Approval

DCP NUMBER 1370-DCP-130

1. DEPARTMENT/SECTION/GROUP APPROVAL(S) COMPLETE.
2. REVIEW COMMITTEE HAS REVIEWED THE DESIGN CHANGE PACKAGE FOR COMPLETENESS AND CONSISTENCY WITH PREVIOUS EVALUATIONS. TECHNICAL POSITIONS TO 10CFR50.59 ARE ADEQUATELY PRESENTED.
3. THERE IS CONCURRENCE WITH THE C-E SITE OFFICE ON THE DETAILS OF IMPLEMENTATION.
4. APPROVALS:

Robert E Keen APPLICATION ENGINEER	1-6-83 DATE	ICE DEPT/SECTION/GROUP
[signature] APPLICATION ENGINEER	1-7-83 DATE	MD DEPT/SECTION/GROUP
W. A. [signature] APPLICATION ENGINEER	1-10-83 DATE	RD DEPT/SECTION/GROUP
[signature] APPLICATION ENGINEER	1-10-83 DATE	PE DEPT/SECTION/GROUP
[signature] SYSTEMS ENGINEERING REPRESENTATIVE	1-11-83 DATE	
[signature] NUCLEAR LICENSING REPRESENTATIVE	7 JAN 83 DATE	
[signature] PROJECT OFFICE REPRESENTATIVE	1/13/83 DATE	
OTHERS (AS SPECIFIED) P.C. Roh	1-7-83 DATE	NSSS Test
	DATE	

IMPLEMENTATION

5. FIELD IMPLEMENTATION OF DESIGN CHANGE (INFORMATION VIA PROJECT OFFICE/SITE)
   A. WORK TO BE PERFORMED BY ____________
   B. CONSTRUCTION SAFETY EVALUATION BY ____________
   C. MATERIALS/COMPONENTS REQUIRED ( )YES ( )NO IF YES, EXPLAIN.
   D. SPARE PARTS AFFECTED. ( )YES ( )NO IF YES, EXPLAIN.
   E. PO/MO NUMBER, IF APPLICABLE ____________

   FIELD IMPLEMENTATION COMPLETED.

   ____________ PROJECT OFFICE ____________ DATE

6. ENGINEERING IMPLEMENTATION COMPLETED.

   ____________ SYSTEMS ENGINEERING ____________ DATE

1.3 GDC-35 Emergency Core Cooling

GDC-35 requires that for the Emergency Core Cooling System, "Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabiliies shall be provided to assure that for on site electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming on site power is not available) the system safety function can be accomplished, assuming a single failure."

A design review of the ECCS has identified single components within the Plant Protection System (connectors AJ3109 and DJ3109) which if disconnected, could cause simultaneous Safety Injection (SIAS) and Recirculation (RAS). In order to state whether the requirements of the GDC are met it must first be determined if disconnection of the connector is a single failure consistent with the intent of IEEE-379 or Regulatory Guide 1.53 which state that the system be capable of performing its function inthe presence of a single failure. This failure is normally attributed to active components not operating when required (e.g., a relay, bistable, circuit breaker, etc.). This could be attributed to failure during component actuation or to prior failure of the component in the interval before periodic surveillance detects the failure.

In the case of the connector (a passive component) the only function it must perform during system actuation is to maintain continuity. To assure that this continuity is maintained the connector is designed with two screws passing through the connector into it's mating half assuring the two halves remain connected. Furthermore, the connector has been seismically qualified to assure that during a seismic event the connector does not disconnect or momentarilly break connection. Also, the PPS cabinet seismic qualification shows that no means exists to break this cable or connector during a seismic event. Therefore, the only means to break this connection is an intentional disconnection. Upon disconnection the operator would have immediate annunciation such that the concern about a failure existing for a short period of time without being identified, is not applicable.

Since the only means for disconnection is through a deliberate act the probability of this occuring simultaneously with an accident is considered incredible. Therefore, this type of failure is considered to be an initiating event rather than a single failure.

Based on the above discussion it can be concluded that the design meets the requirements of GDC-35.

1.4 PPS Tests and Alarms

All vital bus and PPS power supplies are monitored continuously and will annunciate loss of power in the control room. Alarm response procedures direct the operators to initiate corrective action. Therefore, it is extremely unlikely that a seriously degraded power supply could exist undetected or unrepaired for very long.

The following alarms are available to allow the operator to detect degraded power:

Detection of Vital Bus Degraded Power

Vital Bus Inverter System - Alarm Description

Low DC Voltage - An alarm condition will be present if the Inverter DC Input Voltage falls to 110 VDC. At this time the DC Voltage normal light will go out. The alarm will be terminated when the Inverter DC Input Voltage rise above 122 volts and the alarm reset switch is pressed.

Low Air Flow - An alarm condition will be present if the Inverter Air Inlet Filters become restricted with dust or if a fan should fail. At this time the Low Air Flow Light will be lit. The alarm will terminate when the restriction is cleared or the fan is replaced and the alarm reset switch is pressed.

High Inverter Output Voltage - An alarm condition will be present if the Inverter AC Output Voltage rises to 125 volts. At this time the High Inverter Output Voltage light will be lit. The alarm will terminate when the Inverter AC Output voltage falls to 122 volts and the alarm reset switch is pressed.

Inverter Failure - An alarm condition will be present if Inverter AC Output voltage is not present. At this time the Inverter Failure Light will be lit. The alarm will be terminated when the Inverter AC Output Voltage is restored and the alarm reset switch is pressed.

Inverter Overload - An alarm condition will be present if the Inverter Output Current rises to 200 amps or higher for at least 10 msec. At this time the Inverter Overload Light will be lit. The alarm will be terminated when the Inverter Output Current falls below 190 amps and the alarm reset switch is pressed.

The relays used in the above alarm systems are Schrack RM 202610 relays and have a response time of 15 msec. However, this time does not take into account the electronics associated with each alarm.

The DC power supply trouble annunciation is not designed to annunciate degraded power conditions. The power supply trouble annunciation indicates gross loss of DC power supply output to asist the operator to readily localize a power problem within the system.

When a power supply trouble annunciation has been received power has degraded to the level where circuitry in the system has actuated. However, the actuation of this circuitry will not cause a full system actuation since the degradation or loss occurs within one channel.

Monthly surveillance testing of power supplies provides information on power supply performance.

Power Supply Trouble Annunciation

All power supplies in the system provide annuncation when DC output is lost. Annunciator circuits consist of a relay on the output of each supply. The relays drop out on supply failure. Depending on the supply voltage and annunciator relay type, relays drop out between 0.225 and 0.9 volts or greater with a release time between 1.5 and 7.5 msec. The time response of the Plant Annunciator must be added to these figures. Power supply trouble annunciator relays do not seal-in; this function is provided by the Plant Annunciator System.

Matrix Power Supplies

Should vital power to one channel of the PPS be lost, or the loss or degradation of a matrix power supply occur, the condition is detectable by (a) power supply trouble annunciation (b) dropout (extinguishing) of matrix relay indicators, on the matrix test module, (c) dropout (extinguishing) of trip path indicators on the PPS local status panel, the PPS remote reactor trip status panel (in control room), the remote control modules (in the control room), and (d) trip path indicators extinguishing on the ESFAS Auxiliary Relay Cabinet (ARC) control panels. Again no single loss of a vital bus, or power supply within one channel will cause a full system actuation.

Any of the indications listed above will indicate a power supply (vital bus or DC output) problem when the degree of degradation causes circuitry to actuate in the system.

Trip Path Power Supplies

Should vital power to one channel of the PPS be lost, or the loss or degradation of a trip path power supply occur, the condition is detectable by (a) power supply trouble annunciation (b) extinguishing of a single trip path indicator on the PPS local status panel, the PPS remote reactor trip status panel (in the Control Room), the remote control modules (in the Control Room) and (c) trip path indicators extinguishing on the ESFAS ARC control panels.

Again, no single loss of a vital bus, or power supply within one channel will cause a full system actuation.

Any of the indicators listed above will indicate power supply (vital bus or DC output) problems when the degree of degradation causes cicuitry to actuate in the system (e.g., trip path relays de-energized by 3 VDC or greater).

Bistable and Bypass Power Supplies

Should vital power to one channel of the PPS be lost, or the loss or degradation of a bistable or bypass power supply occur the condition is detectable by the power supply trouble annunciation. No other indication or annunciation occurs because bistable and bypass power supplies are auctioneered across two channels. The system will operate normally since no system circuitry, other than power supply annunciation is affected.

Routine surveillance procedures of both vital bus and PPS power supplies involve RMS voltmeter readings so that degraded modes which change RMS output can be detected. It is expected that the on-going task to monitor vital bus power supply performance with regards to noise levels and momentary de-energizations as discussed in SCE's January 14, 1983 letter will provide information to help identify potential sources of chronic noise.

1.5 Regulatory Guide 1.75

The requirements of IEEE-384, 1974 and Regulatory Guide 1.75 were not in effect during the PPS design phase, consequently, they were not used as design basis criteria. Nevertheless, the PPS design is consistent with many aspects of the guides. Section 7.1.2.29 of the San Onofre Units 2 and 3 FSAR discusses the major portions of the guides which are not implemented.

SCE's responses to the NRC questions in the following two areas addressed the NRC's concerns relative to the independence of the redundant PPS channels:

1) Independence of redundant power within the PPS (Response to NRC questions 032.11, 032.18 and 032.32).

2) Isolation capability between redundant safety circuits (Response to NRC questions 032.12 and 032.13).

The response to the above questions demonstrated that the independence of redundant circuits within the PPS has been maintained. In addition, various test reports and analyses were provided in support of the responses to these questions.

1.6 Operation in Technical Specifiction Action Statements

In response to the NRC's verbal request during the January 25, 1983 telephone conversation, the Technical Specifications were reviewed to determine if long term action statements could permit an abnormal condition which might leave the plant in a state that would significantly alter the evaluations of the spurious initiation of ESFAS. Long term action statements were considered to be actions of 72 hours duration and beyond which impact the potential for the occurrence of spurious actuation of ESFAS.

The review addressed two areas of interest. The first involved an assessment of whether or not the action requirements could effect the results of the evaluation of spurious ESFAS transients. The second covered the effect of the action requirements on increasing the potential for the occurrence of spurious actuation of ESFAS.

To address the first issue, long term actions in the technical specifications were identifid and listed in Table 1.6-1. The tablulated items have been reviewed and, in our opinion, none of these long term action statements alter the conclusions of the evaluation of spurious ESFAS transients.

Two Action Statements, related to the ESF portion of the PPS, allow for continued operation for 72 hours and beyond.

Action Statement 9 - Table 3.3-3 of the Technical Specifications

This statement allows the bypass of a PPS bistable (thereby the process measurement circuit) for an unspecified period consistent with Technical Specification 6.5.1.6e. However, if the bistable is not bypassed, it must be placed in a trip condition. Should a failure occur during this time in an identical circuit within another channel, full actuation of the associated ESF would occur.

Action Statement 10 - Table 3.3-3 of the Technical Specifications

This statement is similar to "9", however, if a second failure occurs after the first channel is bypassed, the bistable associated with that failed circuit must be placed in a tripped condition. Upon receipt of a third failure in the identical circuit in another channel, full actuation of that associated ESF would occur.

It should be noted however, that Technical Specification 6.5.1.6e requires that each bypass or retention of a trip condition must be reviewed by the Onsite Review Committee.

Table 1.6-1

SYSTEM TECHNICAL SPECIFICATIONS WITH
LONG TERM ACTION STATEMENTS
(72 hrs. and longer)

TECH. SPEC. #	MODES	OPERABILITY REQUIREMENT
3.1.2.2	1-4	Two boron injection flow paths (charging pumps)
3.1.2.4	1-4	Two charging pumps
3.1.2.6	1-4	Boric acid make-up pump per T/S 3.1.2.2.a.
3.1.2.8a	1-4	Boric acid make-up tank
3.3.4	1-3	Turbine overspeed protection system
3.4.1.2	3	Two RC Loop/RCP
3.4.3	1-3	Two sets of Pressurizer heaters
3.4.5.1	1-4	Containment atmosphere particulate monitoring system and containment atmosphere gaseous radioactivity monitoring
3.4.8.3.1	Mode 4 with any Tc $\leq$235^{o}F, Mode 5, Mode 6 with RV head on	SDCS relief valve/SDCS suction line isolation valves
3.4.8.3.2	Mode 4, all Tc >235^{o}F	See above
3.5.2	1,2,3 (pressurizer pressure >400 psia) & T_{AVG} $\geq$350^{o}F	Two ECCS trains
3.6.1.3	1-4	Containment Airlocks
3.6.1.6	1-4	Containment Structural Integrity
3.6.1.7	1-4	Containment purge supply and exhaust isolation valves
3.6.2.1	1,2,3	Two Containment Spray Trains

Table 1.6-1 (continued)

TECH. SPEC. #	MODES	OPERABILITY REQUIREMENT
3.6.2.2	1,2,3	Iodine Removal System
3.6.2.3	1-4	Two Containment Cooling Trains
3.6.4.1	1,2	Two independent containment hydrogen monitors
3.6.4.2	1,2	Tow independent containment hydrogen recombiner systems
3.6.4.3	1,2	Two independent dome air circulator trains
3.7.1.2	1,2,3	Three auxiliary feedwater pumps
3.7.3	1-4	Two CCW loops
3.7.4	1-4	Two salt water coding loops
3.7.5	1-6	Two control room emergency air cleanup systems
3.7.6	1-4, (5,6)	All snubbers, (sub-systems)
3.7.8.1	1-6	Fire Suppression water system
3.7.8.2	--	Sprinkler system on equipment protection basis
3.7.8.3	--	Fire hose stations on an equipment protection basis
3.7.9	1-6	Fire rated assemblies
3.8.1.1	1-4	AC electric power sources
3.8.4.1	1-4	Containment Penetration Conductor Overcurrent Protective Devices
3.9.1.2	Irradiated Fuel in Storage Pool	Two independent fuel handling buidling post accident cleanup filter systems
3.11.1.3	1-6	Liquid Radwaste Treatment Systems
3.11.2.4	1-6	Gaseous Radwaste Treatment System and Ventilation Exhaust Treatment System

Section 2.0 - Analysis of ESFAS Actuation Scenarios

2.1 Event Descriptions Related to Chapter 15 of the San Onofre Units 2 and 3 FSAR

Introduction

This section provides an evaluation of the following ESFAS during operation in Modes 1 and 2.

1. Emergency feedwater actuation signal (EFAS).

2. Main steam isolation signal (MSIS).

3. Safety injection actuation signal (SIAS).

4. Containment isolation actuation signal (CIAS).

The evaluation presented herein includes combinations of these signals.

This evaluation shows that spurious initiation of these ESFAS does not result in the violation of acceptance criteria for moderate frequency events as applied in Chapter 15 of the FSAR. In addition, the evaluation shows that loss of offsite power on turbine trip in combination with spurious initiation of ESFAS does not result in the violation of acceptance criteria for infrequent events as applied in Chapter 15 of the FASR.

Equipment Actuations

Equipment actuations caused by EFAS, MSIS, SIAS, and CIAS are provided in Tables 2.1-1 through 2.1-4. These tables are complete lists of equipment actuations for EFAS and for MSIS. For SIAS and CIAS, only equipment whose actuation will potentially affect the NSSS response are included.

From Table 2.1-1 it is seen that the main effect of spurious EFAS is the initiation of full auxiliary feedwater (AFW) flow from the two motor driven pumps and from the steam driven pump to the two steam generators. Due to time delays for AFW valves to open, full AFW flow is initiated approximately 50 seconds after EFAS. AFW is pumped from the condensate storage tank which can have a water temperature as low as 40°F. Thus, a spurious ESFAS will increase heat removal from the reactor coolant system (RCS).

From Table 2.1-2 is it seen that the main effect of spurious MSIS is the termination of main feedwater flow to the steam generators and the termination of main steam flow from the steam geneators. Due to the time delay for main feedwater isolation valve (MFIV) closure, main feedwater flow terminates approximately ten seconds after MSIS. Due to the time delay for main steam isolation valve (MSIV) closure, main steam flow terminates approximately five seconds after MSIS. Thus, spurious MSIS during operation at power will cause a decrease of heat removal from the RCS.

From Table 2.1-3 it is seen that the initial effect on the NSSS of SIAS is loss of pressurizer level control due to the closure of the letdown isolation valves in combination with the starting of the backup charging pumps.

From Table 2.1-4 is it seen that the effect on the NSSS of CIAS is similar to the effect of MSIS due to the closure of the MSIVs and MFIVs on CIAS. An additional impact of CIAS is the loss of the pressurizer sprays due to the isolation of instrument air. In addition, due to the termination of component cooling water (CCW) flow to the reactor coolant pumps, procedures for restoration of CCW will be followed.

Evaluation of Spurious ESFAS Combinations

Spurious ESFAS combinations which maximize impact with respect to the following categories during operation in Modes 1 and 2 were chosen.

1. Increase Heat Removal from the RCS.

2. Decreased Heat Removal from the RCS.

3. Increase in RCS Inventory.

A discussion of ESFAS combinations for each of these categories follows.

Increased Heat Removal ESFAS Combinations

Spurious EFAS causes an increase in heat removal from the RCS due to the initiation of full AFW flow to the steam generators. AFW flow will continue until the operator resets EFAS or until the high water level is reached.

If SIAS occurs in combination with EFAS, additional cooling of the RCS will occur due to the addition of low temperature water to the RCS by the charging pumps. None of the other ESFAS in combination with EFAS and/or SIAS will result in additional heat removal from the RCS.

A discussion follows of the ESFAS combination EFAS/SIAS occurring during operation in Modes 1 and 2.

Table 2.1-5a provides a sequence of events for spurious EFAS/SIAS occuring during plant operation at full power. Following initiation of AFW flow, the main feedwater control system will regulate main feedwater flow to maintain steam generator water level. The main effect of AFW initiation will be a reduction of feedwater enthalpy (30 Btu/lbm). The resulting increased heat removal will cause core power to increase by approximately 5% of rated power. The core protection calculators (CPCs) protect against violation of specified acceptable fuel design limits. Spurious EFAS/SIAS at full power results in a less severe heat removal transient than the Increased Main Steam Flow event evaluated in Section 15.1.1.3 of the FSAR. Therefore, spurious EFAS/SIAS at full power will not result in the violation of acceptance criteria for moderate frequency events.

It should be noted that the occurrence of turbine trip with loss of offsite power following spurious EFAS/SIAS will result in a less severe fuel performance transient than is presented in Section 15.1.2.3 of the FSAR.

Table 2.1-5b provides a sequence of events for spurious EFAS/SIAS occurring during plant operation at hot zero power with the reactor critical. Following initiation of AFW flow, steam generator water level will begin to increase. RCS temperatures will decrease until core power increases to match the RCS heat removal rate. Peak core power will reach approximately 8% of rated power. AFW flow will termiate and the core power transient will terminate when the AFW high level shutoff is reached. Spurious EFAS/SIAS at hot zero power critical results in a less severe heat removal transient than the Inadvertent Opening of a Steam Generator Atmospheric Dump Valve evaluated in Section 15.1.1.4 of the FSAR. Therefore, spurious EFAS/SIAS at hot zero power critical will not result in the violation of acceptance criteria for moderate frequency events.

Table 2.1-5c provides a sequence of events for spurious EFAS/SIAS occuring during plant operations during hot zero power with the reactor subcritical. For this case, no heat addition from the core occurs; consequently, RCS cooldown continues (at $10^{\circ}F$/minute) until AFW flow terminates on high steam generator water level. The total RCS temperature reduction is approximately $150^{\circ}F$. This case does not result in an approach to acceptance criteria; it does however, cause normal RCS cooldown rates to be exceeded.

Decreased Heat Removal ESFAS Combinations

A review of Tables 2.1-1 through 2.1-4 indicates that any combination of ESFAS which includes either MSIS or CIAS will cause a decrease in heat removal by the steam generators due to termination of main steam and feedwater flow. The resulting reactor coolant heatup and expansion will dominate the other effects from CIAS and SIAS (i.e., letdown isolation, charging initiation and pressurizer spray termination). In addition, EFAS will be automatically generated subsequent to feedwater termination. Therefore, any combination with MSIS or CIAS will also include EFAS (spuriously or not).

Table 2.1-6 presents the sequence of events for the combination of MSIS, CIAS and SIAS at full power conditions. As noted above the sequence would be largely unchanged for any other combination which includes MSIS or CIAS. Also as indicated by the Table the effect of a spurious EFAS would be to generate the signal at time zero instead of 20 to 60 seconds.

The event scenario defined by MSIS, CIAS and SIAS is nearly identical to that presented for the Loss of Condenser Vacuum in Secton 15.2.2.3 of the FSAR. The only differences between the two are that the FSAR event assumed instantaneous versus gradual termination of steam flow, feed flow, letdown flow and pressurizer sprays. Consequently, the RCS pressurization for spurious ESFAS will be less severe than the FSAR event and will remain within the 2750 psia acceptance criterion. Like the FSAR event, the minimum DNBR will continually increase from the initiation of the spurious ESFAS, and remain above the specified acceptable fuel design limit (SAFDL).

If a loss of offsite power (LOP) is postulated during a spurious MSIS, CIAS and SIAS, the event scenario would include a reactor trip from the Core Protection Calculators (CPCs) due to the coastdown of the reactor coolant pumps. Therefore, if LOP occurred prior to high pressurizer pressure trip, peak RCS pressures would be lower due to the earlier CPC trip. The event sequence would be nearly identical to the Loss of Normal AC Power event presented in Section 15.2.1.4. The CPC trip will ensure that the minimum DNBR remains above the SAFDL.

For LOP occurring after reactor trip, the influence on peak RCS pressure and minimum DNBR would be negligible. The long term NSSS response would again be similar to that in Section 15.2.1.4.

Increasing RCS Inventory ESFAS Combination

Table 2.1-7 presents the sequence of events for the spurious SIAS event with the NSSS at a hot zero power condition. The event definition sequence is the same as that presented in the CVCS Malfunction event in Section 15.5.1.1 of the FSAR, except that the letdown flow is isolated (the FSAR assumed 39 gpm letdown) and the charging flow is highly borated (the FSAR assumed charging and reactor coolant were of equal boron concentration). Concequently, a spurious SIAS will fill the pressurizer at a slightly higher rate. However, if at power, the boration will lower the core power and lower the RCS temperatures thereby contracting the coolant and offsetting the excess charging flow. If at zero power, the contraction will not occur, but the initial pressurizer liquid level is programmed to be lower than at full power. This provides more than 30 minutes for the operator to terminate the excess charging.

Table 2.1-1

Equipment Actuations Caused by EFAS

Equipment	Actuation
Motor driven auxiliary feedwater (AFW) pumps	Start
Steam line isolation valves to steam driven AFW pump turbine	Open
Stop valve to steam driven AFW pump turbine	Opens
Steam generator blowdown valves	Close
AFW pump discharge valves	Open
AFW pump isolation valves	Open

Table 2 .1-2

Equipment Actuations Caused By MSIS

Equipment	Actuation
Main feedwater isolation valves	Close
Main steam isolation valves (MSIVs)	Close
Atmospheric dump valves	Close
Steam generator blowdown valves	Close
Steam generator sample valves	Close
MSIV bypass valves	Close
Main steam line drain valves	Close
Steam line isolation valves to steam driven AFW pump turbine	Close*
AFW pump discharge valves	Close*
AFW pump isolation valves	Close*

* Closure is overridden by EFAS.

Table 2.1-3

Equipment Actuations Caused By SIAS

Equipment	Actuation
Letdown line isolation valves	Close
Low pressure safety injection (LPSI) pumps	Start
LPSI header isolation valves	Open
High pressure safety injection (HPSI) pumps	Start
HPSI header isolation valves	Open
Safety injection tank (SIT) discharge valves	Open
SIT fill line valves	Close
Hot leg injection to reactor coolant drain tank valves	Close
Volume control tank (VCT) outlet valve	Closes
VCT inlet valve	Closes
Boric acid makeup tank outlet valves	Open
Boric acid makeup pump discharge valve	Opens
Boric acid makeup pump recirculation valves	Close
Boric acid makeup pumps	Start
Charging pumps	Start
Diesel generators	Start*

* Does not affect NSSS if offsite power is available.

Table 2.1-4

Equipment Actuations Caused by CIAS

Equipment	Actuation
Main feedwater isolation valves	Close
Main feedwater control valves	Close
Main steam isolation valves	Close
Letdown line isolation valves	Close
RCS sample line valves	Close
Pressurizer sample line valves	Close
Reactor coolant pump seal bleedoff valves	Close
Component cooling water (CCW) inlet valves	Close
CCW return valves	Close
Instrument air line valves	Close*

* Causes loss of pressurizer sprays.

Table 2.1-5a

Sequence of Events for Inadvertent EFAS/SIAS at Hot Full Power

Time (Seconds)	Event	Setpoint or Value
0.0	Spurious EFAS/SIAS.	—
0-60	Backup charging pumps start.	—
	Letdown isolation valves close.	—
	AFW pumps start.	—
	AFW pump discharge and isolation valves open.	—
60-100	The main feedwater control system maintains steam generator water level. (High and low steam generator water level reactor trips protect against excessive steam generator level swings. In addition, main feedwater will ramp back to minimum flow on reactor trip.	—
~100	Peak core power, % of rated power.	~105
	(The CPC low DNBR trip protects against violation of specified acceptable fuel design limits).	
>100	NSSS parameters are stable except for pressurizer water level.	
	Pressurizer water level is slowly increasing. (More than 30 minutes would be required to fill the pressurizer with liquid. The high pressurizer pressure trip protects against overpressurization of the RCS.)	
	The operator can reduce core power and restore the actuated ESF to normal.	—

Table 2.1-5b

Sequence of Events for Inadvertent EFAS/SIAS at Hot Zero Power (critical)

Time (Seconds)	Event	Setpoint or Value
0.0	Spurious EFAS/SIAS.	—
0-60	Backup charging pumps start.	—
	Letdown isolation valve close.	—
	AFW pumps start.	—
	AFW pump discharge and isolation valves open.	—
		—
>60	Steam generator water level begins to increase. RCS cooldown causes core power to increase from zero.	—
~100	Peak core power, % of rated power.	~8
100-600	NSSS parameters are stable except for pressurizer water level and steam generator water level.	
	Pressurizer water level is slowly increasing. (More than 30 minutes would be required to fill the pressurizer with liquid. The high pressurizer pressure trip protects against overpressurization of the RCS).	—
	Steam generator water level is increasing. (The high steam generator water level reactor trip protects against overfilling of the steam generators. Main feedwater will ramp back to minimum flow on reactor trip.)	—

Table 2.1-5c

Sequence of Events for Inadvertent
EFAS/SIAS at Hot Zero Power (subcritical)

Time (Seconds)	Event	Setpoint or Value
0.0	Spurious EFAS/SIAS.	—
0-60	Backup charging pumps start.	—
	Letdown isolation valves close.	—
	AFW pumps start.	—
	AFW pump discharge and isolation valves open.	—
>60	Steam generator water level begins to increase.	—
	RCS cooldown begins at a rate of 10°F/minute.	—
	Pressurizer level decreases because RCS liquid is contracting at approximately six times the rate of charging pump makeup flow.	—
300	Pressurizer empties and reactor trips on low pressurizer pressure.	—
	RCS pressure decreases to HPSI pump shutoff head and remains stable at that value.	—
	Minimum RCS pressure, psia.	1500
	Operator confirms reactor is tripped, confirms full rod insertion, and trips reactor coolant pumps.	—
600	AFW flow terminates on high steam generator water level.	—
	Total decrease of RCS temperature from initial value, °F.	150
>600	RCS temperatures stabilize.	—

Table 2.1-5c (Continued)

>600	RCS pressure increases above HPSI pump shutoff head.
>600	Pressurizer level restored by charging pumps.

Table 2.1-6

Sequence of Events for Inadvertent MSIS/CIAS/SIAS at Hot Full Power

Time (Seconds)	Event	Setpoint or Value
0	Inadvertent MSIS, CIAS, and SIAS generated.	—
	MSIVs, MFIVs, letdown isolation valves and instrument air isolation valves begin to close.	—
	Backup charging pump(s) start and are aligned with the boric acid makeup tanks.	—
	HPSI and LPSI pumps start and with the SITs are aligned to the RCS.	—
5	MSIVs close fully terminating main steam flow.	—
10	MFIVs close fully terminating main feed flow.	—
10-20	High pressurizer pressure trip signal generated, psia.	2382
	Pressurizer safety valves open, psia.	2500
	Steam generator safety valves open, psia.	1100
	Maximum RCS pressure, psia.	<2700
	Minimum DNBR.	>1.19
	Maximum steam generator pressure, psia.	<1150
	Pressurizer safety valves close.	2450
20-60	EFAS generated, percent narrow range.	25
40	Letdown flow is fully isolated.	—
	Instrument air to pressurizer spray control valves is fully isolated - no sprays.	—
10-110	Emergency feedwater reaches the steam generators.	—
120-200	Boron from charging flow reaches the core.	—

Table 2.1-6 (Continued)

500-800	Steam generator safety valves close, psia.	1060
800+	NSSS parameters are stable: gradual cooling until EFAS resets on high steam generator water level, gradual heating up to the steam generator safety valve setpoint and cycling on the valves until EFAS re-initiates. Pressurizer water level slowly increases. (More than 30 minutes would be required to fill the pressurizer with liquid).	—

Table 2.1-7

Sequence of Events for Inadvertent SIAS at Hot Zero Power

Time (Seconds)	Event	Setpoint or Value
0	Inadvertent SIAS generated.	—
	Backup charging pump(s) start and are aligned with the boric acid makeup tanks.	—
	Letdown isolation valves begin to close.	—
	HPSI and LPSI pumps start and with the SITs are aligned to the RCS.	—
40	Letdown flow is fully isolated.	—
50-100	Pressurizer heaters are de-energized and spray flow is initiated, psia.	2275
120-200	Boron from charging flow reaches the core.	—
200+	NSSS parameters are stable, except for pressurizer water level. (More than 30 minutes would be required to fill the pressurizer with liquid).	—

Section 2.2 - Effects of Spurious ESFAS on Specific Systems

An evaluation of plant response following spurious actuation of a combination of ESFAS signals during plant operation in Modes 3-6 has been conducted. The work is presented in two parts. The first part is a detalied review and evaluation of a postulated worst case scenario for operating modes in which the Shutdown Cooling System, including the LTOP system, are in operation. The second part evaluation was done primarily for completeness and covers operating Modes 3 and 4 with RCS cooling via the steam generator(s) alone. These evaluations have been determined to be within the scope of th safety analysis presented in the FSAR.

2.2.1 Shutdown Cooling

Introduction

The plant's response to spurious ESFAS signals during shutdown cooling system operation has been reviewed. For a spurious ESFAS event during shutdown cooling, the principle consideration is the adequacy of the low temperature over pressure protection (LTOP) system in controlling the pressure transients initiated by the signal combinations tabulated on Table 2.2-1. The effect of other signals (e.g., CSAS or CIAS), occuring in conjunction with these combinations does not alter the LTOP transient. The limiting - case ESFAS signal combination during Shutdown Cooling System operation involves SIAS and RAS. The LTOP system ensures that the pressure transient resulting from this combination does not violate the P-T limits given in the Technical Specifications.

The discussion of spurious ESFAS signals during shutdown cooling system (SDCS) operation begins with a review of the applicable plant operating Modes before expanding upon the summary presented in Table 2.2-1. After discussing the various signal combinations of Table 2.2-1, the design of the LTOP system is reviewed. This provides background for the evaluation of the capability of the SDCS relief valve to mitigate and control the limiting-case pressure transient.

SDCS/Plant Operating Modes

Operation of the shutdown cooling system is permissible in Mode 4 with pressurizer pressure less than about 376 psia. A steam generator loop can also be used to remove heat from the Reactor Coolant System (RCS) in this mode. (See Tech. Spec. 3/4.4.1.3.) This discussion applies to plant operation in Mode 4 with the SDCS in operation. With the SDCS in operation, the SDCS relief valve provides protection against overpressure at low temperature. SDCS operation is required in Modes 5 and 6.

Although Technical Specification 3/4.4.1.4.1 allows the steam geneators to serve as a means of standby RCS heat-removal capability in Mode 5, the RCS and steam generator temperatues are nearly in equilibrium in this mode. Thus, an MSIS (or an EFAS) cannot significantly change the heat input to (or heat removal from) the RCS. The steam generator(s) can remove RCS heat in the event shutdown cooling flow is lost and the RCS heats up until the primary to secondary temperature difference favors significant heat transfer to the steam generator(s). This is the bases for steam generator operability requirement in the Technical Specifications.

DETAILED REVIEW OF TABLE 2.2-1

MSIS

With the steam generator(s) and SDCS in operation in Mode 4 and the RCS in water solid condition, a spurious MSIS will decrease the heat transfer from the primary to the secondary system, resulting in a small increase in the RCS pressure due to the thermal expansion of the reactor coolant.

The shutdown cooling system prevents significant heat-up of the RCS and minimizes the pressure change. Opening the mainsteam isolation valve(s) (MSIV) re-establishes steamflow to the steam bypass system.

RAS

RAS stops the LPSI pumps and causes a loss of shutdown cooling flow. This results in a heat-up and expansion of the reactor coolant and an increase in RCS pressure. The RAS pressure transient is enveloped by one of the analysis presented in Section 5.2.2.11 of the FSAR for an inadvertent startup of a Reactor Coolant Pump (RCP) when the RCS is water-solid. The FSAR analysis shows the LTOP system provides adequate protection against overpressure.

The analysis made for the FSAR assumed the steam generator temperature was 100°F greater than the RCS cold leg temperature. This is the limit specified in Technical Specification 3/4.4.1.3. To conservatively maximize the heat input, the analysis also assumed a loss of shutdown cooling. A decay heat load of 1.0% was used. Thus, the inadvertent RCP startup analysis envelopes the RAS event. However, the dominant heat input is due to the 100°F negative temperature difference between the RCS and the steam generator. A loss of shutdown cooling flow produces a mild pressure transient with respect to the FSAR transient.

MSIS-RAS

If the Shutdown Cooling System is in operation with the steam generator(s) in Mode 4, and an RAS is combined with an MSIS, the result will be a decrease in heat removal by the steam generator(s) and a loss of shutdown cooling. The heat-up of the RCS will be more rapid than for a RAS alone and this will result in a more aggressive pressure transient than the one discussed above. However, the MSIS-RAS pressure transient is easily enveloped by the one analyzed for the inadvertent RCP start-up.

SIAS

Section 5.2.2.11 of the FSAR presents the analysis for an SIAS occurring when the LTOP system is operable per Techical Specification 3.4.8.3.1 and 3.4.8.3.2. The SDCS relief valve provides more than adequate protection against overpressurization. This is discussed later in "RAS-SIAS (-MSIS) Pressure Transient."

The spurious SIAS not only pressurizes the RCS, it also causes a cooldown and contraction of the reactor coolant. After the SDCS relief valve opens, flow from the HPSI and charging pumps passes through the core and out the SDCS relief valve, increasing the RCS heat removal rate. The cooling of the reactor coolant tends to redcce the steady-state pressure of the RCS after the relife valve has opened. The FSAR analysis conservatively neglected this pressure reduction.

Depending upon the time after reactor shutdown, it is possible for SIAS to rsult in a pressurization/cooldown transient. When the SDCS is placed in operation in order to finish the cooldown of the plant after reactor shutdown, the flowrate through the shutdown cooling heat exchanger(s) (SDCHX) is limited so as not to exceed the maximum cooldown rate specified in Technical Specification 3/4.4.8.1. HPSI and charging pump flow potentially can increase the RCS cooldown rate requiring operator action to limit cooling by the shutdown cooling system to prevent an excessive cooldown rate.

SIAS-MSIS

If the shutdown cooling system is in operation with the steam generator(s) in Mode 4, and an MSIS is combined with SIAS, the resulting pressure transient is essentially the same as an SIAS alone. This is because the pressurization due to the HPSI and charging pumps is more rapid than the pressurization due to a decrease in steam generator heat removal in Mode 4. The MSIS will decrease the heat transfer from the primary to secondary system, resulting in a small increase in RCS pressure due to the thermal expansion of the reactor coolant. However, the shutdown cooling system will prevent significant heat up of the reactor coolant. In addition, if credit is taken for the cooling effect of the HPSI and charging pump flow, there is no heat up and expansion of the reactor coolant.

SIAS-RAS

This combination of two spurious ESFAS signals in Mode 4 without the steam generators in operation, and in Mode 5, and Mode 6 with the reactor vessel head on produces the limiting-case pressure transient. SIAS dominates the pressure increase from time zero until the SDCS relief valve opens. This already has been pointed out for SIAS-MSIS. The large capacity of the SDCS relief valve can accommodate the HPSI pumps' and charging pumps' flow and relieve the volume expansion of the reactor coolant due to the loss of decay heat removal.

Depending upon the time after reactor shutdown, it is possible for an SIAS-RAS to result in a pressurization/cooldown transient despite the loss of shutdown cooling flow caused by RAS stopping the LPSI pumps. The cooling effect of SIAS has been discussed in the previous subsection. Since the RAS stops the LPSI pumps, the cooldown will be less rapid than for the case of SIAS alone.

SIAS-RAS-MSIS

This combination of three ESFAS signals produces the limiting-case pressure transient in Mode 4 when part of the decay heat load is being removed by the steam generators; (SDCS also is in operation). The SIAS dominates the pressure transient from time zero until the relief valve lifts. Afterward, the loss of heat removal becomes more important. For the potential cooldown case, see the discussion for SIAS.

EFAS

With the steam generators in operation with the SDCS in Mode 4, a spurious EFAS will decrease any plant heatup operation in progress or increase the cooldown rate in a cooldown operation. The cooldown will be within the P-T limits of the Technical Specifications.

SIAS-EFAS

The pressure transient is dominated by the SIAS. Since EFAS increases heat removal by the steam generators when they are in operation in Mode 4, the pressure transient is enveloped by the one for SIAS.

If the cooling effect of the HPSI and charging flows after the SDCS relief valve opens is considered, the increased heat removal with EFAS may require operator action to limit cooling by the shutdown cooling system to prevent an excessive cooldown rate.

SIAS-RAS-EFAS

Since EFAS acts to increase decay heat removal by the steam generators when they are in operation in Mode 4, the pressure transient for this combination is enveloped by the one for SIAS-RAS. The cooldown transient is enveloped by the one for SIAS-EFAS.

Conclusion

A review of the ESFAS signals has shown that the combinations of RAS-SIAS or RAS-SIAS-MSIS have the greatest potential for affecting normal plant operation in Mode 4 with the shutdown cooling system in operation, and mode 5, and Mode 6 with the reactor vessel head on. The principle consideration is the adequacy of the low temperature overpressure protection (LTOP) system in controlling the resulting pressure transient. The discussion which follows develops the conditions associated with these most limiting signal combinations and discusses the response of the LTOP system.

Description of Plant Condition

The shutdown cooling system (SDCS) is in operation with one or both LPSI pumps running. The LPSI pump miniflow recirculation line isolation valves (HV-8162 and -8163) are closed, while the miniflow isolation valves (HV-9306 -9307, -9347, -9348) for the HPSI and containment spray (CS) pumps are open. The reactor coolant system (RCS) is water solid at a pressurizer pressure less than or equal to about 376 psia and the cold leg temperature is less than 235°F. Per Technical Specification 3/4.4.8.3.1 the SDCS relief valve is required for LTOP and all four of the SDCS suction line isolation valves are open. The isolation valve on the letdown line off the RCS cold leg may be open for the purpose of shutdown cooling purification in Mode 4, otherwise it is already closed. If the steam generator(s) is in operation to remove part of the RCS heat in Mode 4, the main steam isolation valve (MSIV) is open and the steam bypass system is in operation. Emergency feedwater supplies the steam generator.

Sequence of Events and Systems Operation

There is a spurious initiation of SIAS and RAS. The SIAS starts both HPSI pumps, both LPSI pumps (if one LPSI pump was not running for shutdown cooling purposes), both CS pumps, and closes the letdown isolation valve. The injection of mass into the RCS by the HPSI pumps begins the pressure transient.

The RAS stops the LPSI pumps and opens the isolation valves in the recirculation line from the containment sump. Stopping the LPSI pumps results in a loss of shutdown cooling, causing a heat-up and expansion of the reactor coolant, which contributes to the pressure transient. whether or not the miniflow valves are open, the HPSI pumps are not dead-headed because the SDCS relief valve passes all of the pumps' flow to the containment sump. With the containment at normal operating pressure, opening the sump isolation valves does not affect ECCS pump suction. There is sufficient elevation head from the refueling water storage tank to provide pump suction. Check valves prevent the RWST from draining into containment sump.

For RAS-SIAS-MSIS, the MSIS closes the MSIV. Steam generator pressure increases. Heat removal from the RCS decreases.

The required actions to recover from a spurious SIAS-RAS during shutdown cooling are to immediately restart the LPSI pumps and then stop the HPSI and charging pumps (and the CS pumps which have been started by the SIAS). Per the LTOP design requirements, no operator action is credited for 10 minutes following the event.

The prime consideration in the event of a spurious RAS-SIAS or RAS-MSIS-SIAS at low temperature is the ability of the LTOP system to mitigate the resulting pressure transient. The ability of the LTOP system to prevent an overpressurization of the RCS and SDCS has been reviewed. The results are summarized in the next section.

Summary of Results

The as-built Low Temperature Overpressure Protection (LTOP) System can accommodate a scenario involving a spurious RAS-SIAS or RAS-SIAS-MSIS as the initiators of a potential overpressure event. This capability is available in the existing LTOP system because the relief valve and discharge piping installed at San Onofre Units 2 and 3 have a flow capacity that is much larger than the LTOP design requirements. The design required flow capacity of the limiting-case single failure scenarios which potentially can initiate an overpressure event is approximately 1600 gpm. The installed relief valve has a flow capacity of 3089 gpm, which can accommodate an overpressure event involving a spurious RAS and SIAS (and MSIS).

For this spurious actuation scenario, the pressure transient will not exceed 110% of the design pressure of the shutdown cooling system (SDCS). The reactor vessel and reactor coolant system (RCS) are protected against overpressurization at low temperature.

The following sections discuss the LTOP system and the pressure transients expected from a spurious SIAS-RAS (-MSIS).

LTOP System Design Requirements

The LTOP system design meets the staff requirements on protection against overpressure during water solid operation at low temperature, as decribed in the response to NRC Questions 212.117 and 212.141. As noted in FSAR Section 5.2.2.11.1.2, the design protects the reactor vessel and RCS against over pressure at low temperature given a single failure in addition to the failure which initiates the pressure transient. The design requirements are discussedd in FSAR Sections 3.2.1, 3.9.3.3 and 7.6.1.1, 7.6.2.1. The interlocks on the SDCS suction line isolation valves specified in the LTOP Tech. Spec. 3/4.4.8.3.1 are not controlled by the Engineered Safety Features Actuation System. The following section discusses the performance of the LTOP system in preventing overpressurization at low temperatures.

Design Transients

The event initiating the pressure transient was considered to result from either an operator error or an equipment malfunction (single failure). The varous events considered in the FSAR analysis are characterized by either a mass or an energy input to the reactor coolant system.

There are two limiting-case overpressure transients, both resulting from operator error: 1) A spurious SIAS which activates both HPSI pumps and all three charging pumps, (mass input); or 2) An inadvertent start-up of a reactor coolant pump (RCP) with the RCS water solid and with heat input to the RCS from the steam generator due to a negative temperature difference of $100^{o}F$ (this is the maximum temperature difference permitted by Technical Specification 3.4.1.3).

To maximize the heat inputs for the analysis of the RCP startup, a loss of shutdown cooling flow was assumed.

Loss of LPSI pump(s) results in a loss of decay heat removal and pressurization of the water solid RCS. Spurious startup of all pressurizer heaters is not the limiting single failure of the equipment or control systems. Pressurizer heat input is only 1500 BTU/SEC. Conservatively, assuming a decay heat rate of 1.0% of the rated core power, the heat input to the RCS on loss of shutdown cooling is 32,200 BTU/SEC.

RAS-SIAS (-MSIS) Pressure Transient

The RAS-SIAS (-MSIS) pressure transient is enveloped by a combination of mass input and energy input events. The mass input event is equivalent to the FSAR analysis for the startup of both HPSI pumps, all the charging pumps and the isolation of the letdown line on SIAS with the RCS water solid. As shown in Figure 2.2-1 (FSAR Figure 5.2-2E), the SDCS relief valve provides ample protection against over pressurization. The relief valve's steady state discharge flow rate is approximately 1600 gpm for this transient. The same figure also shows the pressure transient for the limiting energy input case (RCP startup). The SDCS relief valve provides ample protection against overpressure. The relief valve's steady state discharge flowrate is less than the 1600 gpm flowrate for SIAS.

The rated capacity of the relief valve is 3089 gpm at its accumulation pressure. The installed valve and discharge piping are oversized for the design transients presented in the FSAR. The relief valve is capable of handling both worst case mass and energy input transients simultaneously.

In the section which discussed the MSIS-RAS combination it was noted that the pressure transient for the FSAR analysis of an inadvertent RCP startup easily enveloped the one for the MSIS-RAS combination. The corresponding relief valve flowrate needed to accommodate the expansion of the reactor coolant during SIAS-RAS (-MSIS) is much less than the flow requirement for the RCP startup. If there is a spurious actuation of SIAS-RAS (-MSIS), the resulting pressure transient will produce a steady state discharge flowrate from the relief valve which is larger than the flowrate for an SIAS alone. However, the required discharge flowrate at the peak of the RAS-SIAS (-MSIS) transient is less than the capacity of the relief valve. The LTOP system's relief valve will provide the necessary overpressure protection for a spurious actuation of SIAS and RAS (and MSIS) at low temperatures.

TABLE 2.2-1

SUMMARY OF LIMITING TRANSIENTS
INITIATED BY SPURIOUS ESFAS SIGNAL(S)
DURING SHUTDOWN COOLING SYSTEM OPERATION[1]

ESFAS SIGNAL(S)	TYPE OF EVENT	ENVELOPING ANALYSIS	EVALUATED IN FSAR?	DOES SDCS RELIEF VALVE PROVIDE ADEQUATE PROTECTION?	COMMENTS
MSIS[2]	HEAT-UP	OPERATOR ERROR: STARTUP OF AN RCP WITH A WATER-SOLID RCS AND A SECONDARY TO PRIMARY SYSTEM $\Delta T = 100^{o}F$. LETDOWN IS ISOLATED AND DECAY HEAT IS 1.0% OF RATED CORE POWER.	YES	YES	THE MSIS TRANSIENT DOES NOT APPROACH THE SEVERITY OF THE FSAR ANALYSIS TRANSIENT. THE SHUTDOWN COOLING SYSTEM PICKS UP THE EXTRA HEAT LOAD, REDUCING THE PRESSURE INCREASE.
RAS	HEAT-UP	SEE ABOVE	YES	YES	RESTORING A LPSI PUMP RESTORES SHUTDOWN COOLING.
MSIS[2]-RAS	HEAT-UP	SEE ABOVE	YES	YES	SEE ABOVE COMMENTS.
SIAS	PRESSURIZATION / COOLDOWN[3]	MANUAL INITIATION OF SIAS STARTS BOTH HPSI PUMPS, AND THREE CHARGING PUMPS WITH THE RCS WATER SOLID AND LETDOWN ISOLATED.	YES	YES	THE FSAR ANALYSIS IN SECTION 5.2.2.11 CONSERVATIVELY NEGLECTS THE PRESSURE REDUCTION DUE TO THE COOLING EFFECT OF THE HPSI PUMP FLOW THROUGH THE CORE AFTER THE SDCS RELIEF VALVE OPENS. THIS MAXIMIZES THE STEADY-STATE PRESSURE. STOPPING THE HPSI AND CHARGING PUMPS TERMINATES THE TRANSIENT.
SIAS-MSIS[2]	PRESSURIZATION / COOLDOWN[3]	SEE ABOVE	YES	YES	THE PRESSURE TRANSIENT IS ESSENTIALLY THE SAME AS SIAS ALONE. THE MSIS MODERATES THE COOLDOWN AND INCREASES THE PRESSURE SLIGHTLY. STOPPING THE HPSI AND CHARGING PUMPS TERMINATES THE TRANSIENT. OPENING THE MSIV REESTABLISHES STEAM FLOW.

TABLE 2.2-1 (Continued)

SUMMARY OF LIMITING TRANSIENTS
INITIATED BY SPURIOUS ESFAS SIGNAL(S)
DURING SHUTDOWN COOLING SYSTEM OPERATION[1]

ESFAS SIGNAL(S)	TYPE OF EVENT	ENVELOPING ANALYSIS	EVALUATED IN FSAR?	DOES SDCS RELIEF VALVE PROVIDE ADEQUATE PROTECTION?	COMMENTS
SIAS-RAS	PRESSURIZATION / HEAT-UP OR[3] PRESSURIZATION / COOLDOWN	SEE DISCUSSION	HEATUP: NO COOLDOWN: YES	YES[4] YES	FOR THE HEAT-UP CASE, HPSI PUMP FLOW REMOVES CORE HEAT SO THAT THE SDCS DESIGN TEMPERATURE IS NOT EXCEEDED. RESTARTING THE LPSI PUMP RESTORES SHUTDOWN COOLING. STOPPING THE HPSI AND CHARGING PUMPS TERMINATES THE PRESSURE TRANSIENT. FOR THE COOLDOWN CASE, STOPPING THE HPSI AND CHARGING PUMPS TERMINATES THE PRESSURE TRANSIENT AS WELL AS THE COOLDOWN.
SIAS-RAS-MSIS[2]	SEE ABOVE	SEE ABOVE	SEE ABOVE	SEE ABOVE	SEE ABOVE TWO COMMENTS
EFAS[2]	COOLDOWN	—	—	YES	COOLDOWN IS WITHIN P-T LIMITS — NO ADVERSE IMPACT,
SIAS-EFAS[2]	PRESSURIZATION / COOLDOWN[3]	SEE SIAS	SEE SIAS	SEE SIAS	SEE SIAS
SIAS-RAS-EFAS[2]	SEE SIAS-RAS	SEE SIAS-RAS	SEE SIAS-RAS	SEE SIAS-RAS	EFAS REDUCES THE HEAT-UP CASE. THE LIMITING HEAT-UP CASE IS SIAS-RAS. ALSO SEE SIAS-RAS COMMENTS.

TABLE 2.2 (Continued)

SUMMARY OF LIMITING TRANSIENTS
INITIATED BY SPURIOUS ESFAS SIGNAL(S)
DURING SHUTDOWN COOLING SYSTEM OPERATION[1]

ESFAS SIGNAL(S)	TYPE OF EVENT	ENVELOPING ANALYSIS	EVALUATED IN FSAR?	DOES SDCS RELIEF VALVE PROVIDE ADEQUATE PROTECTION?	COMMENTS

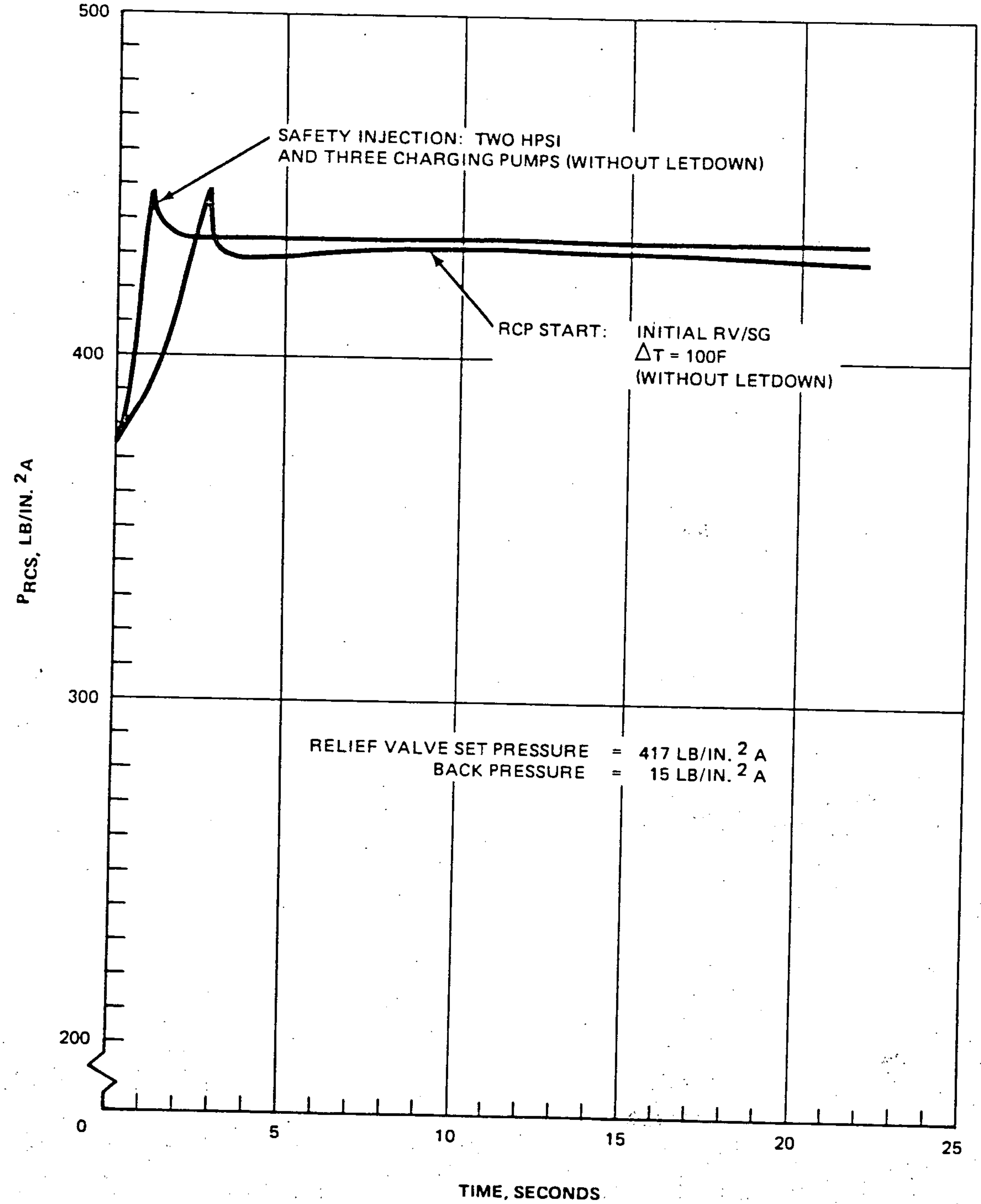
NOTES: (1) MODES 4 AND 5 AND MODE 6 WITH THE REACTOR VESSEL HEAD ON. TECHNICAL SPECIFICATION 3.4.8.3.1 REQUIRES THE OPERABILITY OF THE SDCS RELIEF VALVE AT LOW TEMPERATURE TO PROVIDE LTOP.

(2) MODE 4 ONLY. MSIS AND EFAS HAVE NO IMPACT IN MODES 5 AND 6.

(3) HEAT-UP/COOLDOWN DEPENDS ON THE TEMPERATURE OF THE RWST WATER INJECTED BY THE HPSI PUMP AND THE TIME AFTER REACTOR SHUTDOWN. IN MODE 4, AN SIAS ALWAYS RESULTS IN A PRESSURIZATION/COOLDOWN TRANSIENT.

(4) THE LARGE CAPACITY OF THE SDCS RELIEF VALVE HAS RESERVE CAPACITY TO MITIGATE AND CONTROL THE PRESSURE TRANSIENT. SEE DISCUSSION.

Figure 2.2-1



SAN ONOFRE
NUCLEAR GENERATING STATION
Units 2 & 3

LIMITING TRANSIENTS WITH
SDC RELIEF VALVE PROTECTION

Figure 5.2-2E

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27JA8

2.2.2 Steam Generator Cooling

Introduction

An evaluation of plant response following spurious actuation of a combination of ESFAS signals during plant operating Modes 3 and 4 (RCS cooling via steam generators only) has indicated that these events are enveloped by the design basis events in the FSAR.

The combinations of ESFAS signals reviewed are tabulated on Table 2.2-2. Other combinations of ESFAS signals not presented in the table were considered but are enveloped by the combinations reviewed in detail herein. Table 2.2-2 illustrates the type of event expected for each of the signal(s) actuated and notes the reference which addresses the type of event postulated. The limiting event in this review is the combination SIAS-MSIS-CIAS.

Operating Modes

Modes 3 and 4 are hot, subcritical modes of operation covering RCS temperatures greater than 200°F (Technical Specification Table 1.1 - Operational Modes). This review assumes that RCS cooling is being effected using at least one RCS loop via a steam generator. The RCS pressure-temperature (P-T) relationship is consistent with RCP operation. For Mode 3 (Hot Standby), the upper P-T limits are the same as the Mode 1/2 limits and present no more severe circumstances than any of the Mode 1/2 cases which are covered in the review section for spurious ESFAS signals during power operation. The P-T situation that makes Mode 3/4 distinct from Mode 1/2 is the case of the lowest RCS pressure (minimum subcooling for RCP operation) such that safety injection flow will be maximized at the onset of the event, and the lowest RCS temperature such that the potential for the maximum RCS expansion occurs. The limiting case transient couples loss of heat removal (energy input), increased RCS inventory (mass input), and loss of or diminished RCS pressure control.

The spurious actuation of SIAS, initiates safety injection from the HPSI pumps at a rate consistent with delivery curves. The injection rate at 400 psia will be large for two HPSI pumps, but will decrease with increasing RCS pressure up to the shutoff head of approximately 1200 psia. LPSI pumps will start but the shutoff head for LPSI is about 200 psia and so it will not increase RCS inventory. Obviously, the actuation of RAS would have no bearing on this scenario. All non running charging pumps will start and letdown will be isolated. The cold water injection would tend to contract the RCS opposing the pressurizer level increase but the dominant effect will be the mass increase causing an increase in pressurizer level. The mass increase will behave in manner similar to the event CVCS Malfunction in Chapter 15 of the FSAR with the exception of a Reactor Trip since the reactor is already shutdown.

A Main Steam Isolation Signal (MSIS) will shut the Main Steam Isolation Valves, the Atmospheric Dump Valves, the Main Feed Isolation Valves, and the Auxiliary Feedwater Isolation Valves. The MSIS results in an RCS heat up if decay heat is present which causes expansion of the RCS, further incresing pressurizer level.

RCS pressure increases as pressurizer level increases. Pressure will increase quickly to the HPSI shutoff head which terminates safety injection flow. Pressurizer level and pressure will continue to increase as the RCS heats up and charging flow continues. Assuming no operator actions, two possibilities exist according to the heatup rate and initial RCS temperature. The RCS pressure may increase to the pressurizer safety setpoint followed by a heatup to the saturation temperature for the secondary safety valves pressure setpoint or the RCS temperature may cause the secondary safeties to lift prior to lifting primary safeties. In either case RCS heat removal will be established through the secondary safety valves which will cooldown the RCS. Either scenario is enveloped by the Loss of Condenser Vacuum, FSAR chapter 15.2.1.3.

A Containment Isolation Actuation Signal will cause a loss of cooling water to the reactor coolant pumps. Reactor operators are trained to recover cooling water to the RCPs following CIAS, however, assuming the RCPs trip was required, the event will still be enveloped by the Loss of Condenser Vacuum analysis although the final RCS pressure may peak higher than the MSIS-SIAS scenario alone due to a delay in establishment of RCS flow by natural circulation. The heat generated by RCPs will be a significant portion of total RCS heat load for any decay heat assumed, given that, the addition of the CIAS may mitigate the transient. CIAS also isolates instrument air to the pressurizer main spray valves. Spray may be actuated by auxiliary spray or after a CIAS reset signal.

Mitigation of this type of event will involve termination of the Safety Injection System, reset of the CIAS for RCP cooling and pressurizer spray valve control and establishing RCS heat removal.

RCS heat removal may be satisfied by reopening the Main Steam Isolation Valves and steaming via the turbine bypass valves (TBVs) or bleeding steam through the Atmospheric Dump Valves (ADVs). These two success paths for heat removal can be expected to control decay heat removal almost immediately after clearing the MSIS. Heat removal will moderate any pressure trainsient that ensues by minimizing or counteracting coolant expansion due to heatup.

If RCPs lose power, RCP coastdown will allow for coolant transport temporarily. Following coastdown, conditions for natural circulation thermal driving head will develop. This may cause a short term heat up.

The rate of pressurizer level increase is dependent on the RCS pressure response from the competing effects of mass input (inventory increase) and energy removal (inventory and pressure decrease). This situation is easily diagnosed and corrected since the HPSI pump termination criteria will be satisfied during the course of the transient. The operators need only terminate safety injection.

Operator action consistent with existing operating instructions is required for re-establishing decay heat removal via the SG's by clearing the MSIS and modulating the TBV's or ADV's, regaining control of pressurizer spray by restoring instrument air to the main spray valves and terminating safety injection.

Each safety function affected can be attended to in a timely manner by reasonable operator actions consistent with exsiting operating instructions. No deleterious consequences are anticipated for the scenario described.

TABLE [illegible].2-2

MODE 3/4 WITH STEAM GENERATOR COOLING

ESFAS SIGNAL(S)	TYPE OF EVENT	ENVELOPING ANALYSIS	EVALUATED IN FSAR?	WITHIN ANALYTICAL ENVELOPE?	COMMENTS
SIAS	PRESSURIZATION / COOLDOWN	HPSI SELF-LIMITING ON PUMP SHUTOFF HEAD. CHARGING PUMP INJECTION COVERED IN FSAR SECTION 15.5	YES	YES	THE COOLDOWN IS LIMITED BY THE DECREASE IN CORE OUTLET TEMPERATURE WHICH REDUCES RCS SG HEAT TRANSFER.
SIAS-EFAS	PRESSURIZATION / COOLDOWN	SAME AS ABOVE, POSSIBLE HIGHER COOLDOWN RATE	YES	YES	SAME AS ABOVE, AUXILIARY FEED WILL ALREADY BE IN USE FOR MODES 3/4.
EFAS	POSSIBLE SMALL COOLDOWN	WITHIN OVERCOOLING TRANSIENT SCENARIOS; SEE FSAR SECTION 15.1.1.2, "INCREASE IN FEEDWATER FLOW"	YES	YES	INCREASED STEAM GENERATOR INVENTORY POSSIBLE.
MSIS-EFAS	HEAT-UP	DECREASE IN HEAT REMOVAL FSAR SECTION 15.2.1.3, "LOSS OF CONDENSER VACUUM"	YES	YES	SAME AS MSIS BUT MODERATED BY POSSIBILITY OF INITIAL INCREASED AUXILIARY FEED FLOW.
MSIS	HEAT-UP	DECRASE IN HEAT REMOVAL FSAR SECTION 15.2.1.3	YES	YES	SG INVENTORY ADEQUATE FOR RCS COOLING THROUGH SECONDARY SAFETY VALVES.
SIAS-MSIS	PRESSURIZATION / HEAT-UP	SEE CIAS	YES	YES	LOSS OF PRESSURIZER LEVEL CONTROL
SIAS-CIAS	PRESSURIZATION / HEAT-UP	SEE CIAS	YES	YES	LOSS OF PRESSURIZER LEVEL/PRESSURE CONTROL.

TABLE 2.2-2 (Continued)

MODE 3/4 WITH STEAM GENERATOR COOLING

ESFAS SIGNAL(S)	TYPE OF EVENT	ENVELOPING ANALYSIS	EVALUATED IN FSAR?	WITHIN ANALYTICAL ENVELOPE?	COMMENTS
CIAS	HEAT-UP	"LOSS OF CONDENSER VACUUM WITH A CONCURRENT SINGLE FAILURE" FSAR SECTION 15.2.2.3	YES	YES	UNLIKELY BUT POSSIBLE LOSS OF FORCED CIRCULATION.
MSIS-CIAS	HEAT-UP	SEE MSIS AND CIAS SECTIONS	YES	YES	SEE MSIS AND CIAS SECTIONS.
SIAS-MSIS-CIAS	SEE DISCUSSION	LOSS OF HEAT REMOVAL, ASSUMED LOSS OF FORCED CIRCULATION, INCREASE IN RCS INVENTORY	YES	YES	WORST CASE SCENARIO

2.2.3 Load Shedding

The SIAS is also used as the signal to open certain breakers to isolate their loads from the ESF buses during a LOCA. This design basis is described in Sectin 8.3 of the FSAR and recently has been further analyzed in response to NRC question 222.42 (Amendment 23) relative to the requirements of IE Bulletin 80-06. A summary follows.

SIAS starts the diesel generators, and starts the time delay relays which sequence the safety equipment loads onto the ESF buses. (It should be noted that the loss of voltage (LOV) logic for the ESF buses is completely separate from the PPS, and determines whether the bus should be cleared and then powered by the diesel generator prior to load sequencing.) The FSAR Chapter 15 accident analyses and the Technical Specification response times are based on the longest time delay associated with the LOP/SIAS event.

Consequently, the effects of spurious SIAS that are not addressed in the analyses performed in Section 2.1 of this report are limited to that equipment which is separated from the ESF buses. The only safety loads that trip on SIAS are the auxiliary feedwater (AFW) pumps (P140 and P141). The presence of an EFAS for any reason will automatically resequence these pumps onto their ESF buses consistent with accident analysis assumptions. If the spurious SIAS occurs at hot standby when the pumps are in manual operation, the operator overrides the signal and restarts the pumps; if the steam generator level were to reach the EFAS setpoint, the pumps automatically start.

The remainder of the loads whose breakers are tripped by SIAS are non-1E or non-essential and are separated consistent with Regulatory Guide 1.75 recommendations. Each is equipped with an SIAS override to allow manual reconnection if desired. Spurious separation of these loads does not produce any significant plant transient nor interfere with the execution of operating procedures. The loads are:

Pressurizer Backup Heaters E-128 and E-129
Health Physics Computer*
Technical Support Center*
Fire Detection and Actuation System
Essential Lighting System
Low Pressure Turbine Emergency Spray Water Pump
UHF Radio System
Essential Plant Parameters Monitoring Panel

* These loads continue to receive power through a non-1E uninterruptable power supply (UPS).

2.2.4 Reactor Coolant Pumps

The CIAS function of isolating the containment results in closing the supply and return valves for the non-critical component cooling water (CCW) to the reactor coolant pumps (RCPs). A spurious CIAS can initiate this event (loss of CCW to the RCPs) but this is a previously defined scenario from which the operators have been trained to rapidly recover. The RCP seals are designed to withstand loss of CCW for 3 to 5 minutes with no damage and have been tested to operate up to 30 minutes with no significant failure (controlled bleedoff flow did not exceed 2 gpm) (refer to FSAR responses to NRC questions 10.13, 10.29, 10.48, 212.159 and 10.70). This event is considered to impact equipment only. The operating instructions for recovery from the loss of CCW to the RCPs, turbine trip and loss of load are included for NRC information in Section 2.2.5.

2.2.5

The following operating instructions for San Onofre Units 2 and 3 are enclosed:

Loss of Reactor Coolant Pump Component Cooling Water
Loss of Load/Reactor-Generator Mismatch
Emergency Plant Shutdown

N05

SAN ONOFRE NUCLEAR GENERATING STATION
UNITS 2 AND 3

OPERATING INSTRUCTION SO23-3-5.26.1
EMERGENCY
REVISION 4

EFFECTIVE DATE DEC 24 1982

LOSS OF REACTOR COOLANT PUMP COMPONENT COOLING WATER

TABLE OF CONTENTS

SITE FILE COPY

PAGES CHANGED WITH THIS REVISION: 1-5; Attachment 1 - p.2;
Attachment 2 - pps. 1-2;
Attachment 3 - p.1;
Attachment 4 - p.1

PREPARED BY: Robert H Clenat 12-23-82
PROCEDURE WRITER DATE

APPROVED BY: JR Tate for H E Morgan 12/23/82
H. E. MORGAN DATE
MANAGER, OPERATIONS

NUS:230007

LOSS OF REACTOR COOLANT PUMP COMPONENT COOLING WATER

1.0 SYMPTOMS

1.1 Alarms.

1.1.1 CCW From RCP Seal Heat Exchanger Temperature Hi.

1.1.2 CCW From RCP Flow Hi/Lo.

1.1.3 CCW Surge Tank T003 (T004) Level Lo-Lo

1.1.4 CCW Non-Critical Loop Return Flow Lo

1.2 Plant Monitoring System Alarms.

1.2.1 CCW From RCP Temp. High.

1.2.2 RCP Controlled Bleed Off Temp. High.

1.2.3 RCP Stator Temp High.

1.2.4 RCP Air Clr A(B) Air Outlet Temp High.

1.2.5 RCP Lube Oil Clr A(B) Outlet Temp High.

1.3 Indications.

1.3.1 Individual RCP middle, upper and vapor seal cavity pressure indications erratic.

1.3.2 Individual RCP controlled bleed off temperature indications above normal.

2.0 AUTOMATIC ACTION

2.1 Reactor Coolant Pump seal heat exchanger CCW return initiates alarms on CCW high temperature. (TI-9144, TI-9154, TI-9164, TI-9174) R

2.2 CCW non-critical loop supply isolations, HV-6212 and HV-6213, close on Lo-Lo level in the CCW surge tank in the related train.

3.0 IMMEDIATE OPERATOR ACTION

3.1 If CCW cannot be restored to a RCP motor oil cooler within three (3) minutes, or to a RCP seal heat exchanger within five (5) minutes, then complete the following:

3.0 IMMEDIATE OPERATOR ACTION (Cont'd)

3.1.1 In Modes 1 and 2 (power operation and startup), trip the reactor and the turbine. After the CEA bottom lights have been on greater than 5 seconds, stop the affected RCP(s). |D

3.1.2 In all other modes, stop the affected RCP(s). |D

4.0 SUBSEQUENT OPERATOR ACTION INITIALS*

NOTE: Anticipated Event Senarios

(1) The most probable cause of Loss of CCW to all RCPs is automatic isolation of the Non-Critical Loop and/or closure of the Non-Critical Loop containment isolation valves due to CIAS (and/or SIAS Unit 3). |D

Actions are taken to determine the cause of the loss of CCW. Then, specific recovery actions are taken to restore CCW flow and/or minimize Reactor Coolant Pump damage.

4.1 Verify that all immediate operator actions have been initiated as follows:

4.1.1 If the CCW supply to the Non-Critical Loop has been lost or isolated and the other loop is available, then perform Attachment 1, "Recovery of Non-Critical Loop from a Single Train CCW Casualty." ________

* The INITIAL column is an operator aid and is intended to be used as follows:

Initial each completed action. Do not write N/A. Leave blank, items that are not applicable. Proceed through the instruction performing all applicable steps frequently rechecking those steps passed over to ensure action is taken when applicable.

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.2 If the CCW supply to the Non-Critical Loop has been lost or isolated and the other loop is not available, then perform Attachment 2, "Complete Loss of the Non-Critical Loop." ______

4.1.3 If the Non-Critical Loop supply has not been lost or isolated, and the Non-Critical Loop containment isolation valves have not closed, then perform Attachment 3, "Loss of CCW to a Single RCP." ______

4.2 If RCP(s) were being operated as part of, or following, a RCS filling and venting operation, then verify CCW to RCPs valve lineup is correct. ______

4.3 Verify the RCP Seal Heat Exchanger CCW return valve(s) is(are) functioning properly. ______

RCP/Valve	P001/TV-9144	P002/TV-9174
	P003/TV-9154	P004/TV-9164

4.3.1 If necessary, then have Instrumentation and Control perform a circuit verification to ensure the valve is functioning properly. ______

4.4 Refer to Technical Specification 3.4.1.1 (Modes 1 & 2), 3.4.1.2 (Mode 3), or 3.4.1.3 (Mode 4) for limiting conditions for operation. ______

4.5 If maintenance is required, then notify maintenance to initiate repairs. ______

4.6 The Shift Supervisor shall notify the "duty" Station Administrator and Shift Technical Advisor and discuss the situation. ______ R

4.6.1 An assessment of the plant status and safety shall be made and the event classified per SO123-VIII-11, "Recognition and Classification of Emergencies." ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.6.2 If an emergency is declared (Unusal, Alert, Site Emergency or General Emergency), use the following Emergency Procedures to implement the SONGS 2 & 3 Emergency Plan: R

4.6.2.1 Unusual Event - S0123-VIII-12. ________

4.6.2.2 Alert - S0123-VIII-13. ________

4.6.2.3 Site Emergency - S0123-VIII-14. ________ R

4.6.2.4 General Emergency - S0123-VIII-15. ________ R

4.6.3 If unable to contact any Station Administrator in the normal reporting chain within one (1) hour following the declared emergency, then notify the NRC via the red phone. ________ R

5.0 ATTACHMENTS

5.1 Attachment 1, "Recovery of Non-Critical Loop from a Single Train CCW Casualty" (3 pages)

5.2 Attachment 2, "Complete Loss of the Non-Critical Loop" (4 pages)

5.3 Attachment 3, "Loss of CCW to a Single RCP" (1 page)

5.4 Attachment 4, "Failure to Restore CCW" (1 page)

6.0 REFERENCES

6.1 Not Applicable.

7.0 RECORDS

7.1 Upon termination of this procedure and applicable attachments, file in the Operations Evolution File.

7.2 Completed procedure and attachment shall be transmitted to CDM for retention and storage in accordance with applicable station procedures. 2

D

RECOVERY OF NON-CRITICAL LOOP FROM A SINGLE TRAIN CCW CASUALTY

*INITIALS

1.0 Attempt to restore CCW to the non-critical loop within three (3) minutes by completing the applicable section of this attachment. ______

1.1 If unable to restore CCW within three (3) minutes, then complete Attachment 4, "Failure to Restore CCW." ______

2.0 If Train A CCW has been lost and Train B is available, then perform the following: ______

2.1 Close supply to radwaste header HV-6465. ______

2.2 Check closed return from Train B containment emergency coolers HV-6368 and HV-6373. ______

2.3 Open non-critical loop supply and return Train B valves HV-6213 and HV-6219. ______

3.0 If Train B CCW has been lost and Train A is available, then perform the following:

3.1 Close supply to radwaste header HV-6465. ______

3.2 Check closed return from Train A containment emergency coolers HV-6367 and HV-6371. ______

3.3 Open non-critical loop supply and return Train A valves HV-6212 and HV-6218. ______

NOTE: Return CCW temperature from the RCP seal heat exchangers will rise upon restoration of flow.

4.0 Restore CCW flow to the non-critical loop as follows:

4.1 Override and open non-critical loop containment isolation valves HV-6211 and HV-6216. ______

4.2 If any RCP seal heat exchanger CCW return valve (TV-9144, 9154, 9164 or 9174) closes upon restoration of flow, then attempt to promptly open the valve. ______

* The INITIAL column is an operator aid and is intended to be used as follows:

Initial each completed action. Do not write N/A. Leave blank, items that are not applicable. Proceed thorugh the instruction performing all applicable steps frequently checking those steps passed over to ensure action is taken when applicable.

4.0 (CONTINUED)

INITIALS

4.2.1 Monitor RCP seal heat exchanger outlet temperatures on TI-9144, 9154, 9164 and 9174. If CCW temperature from the seal heat exchanger does not stabilize below 250°F, then close the affected RCPs seal heat exchanger outlet valve and proceed per Attachment 4, "Failure to Restore CCW." ______

4.3 Monitor the CCW loop flow on FI-6243 for Train A for FI-6248 for Train B. If flow exceeds 14,000 gpm on the loop supplying the non-critical loop, then consider isolating CCW supply valves to SFP heat exchangers, HCV-6545 and HCV-6546. ______

4.4 Verify adequate flow has been restored to the RCPs as indicated by the following alarms clearing:

4.4.1 RCP P001(2, 3 or 4) SEAL PRESS HI/LO 56C24, 56C26, 56C28, 56C30. ______

4.4.2 CCW FROM RCP P001 (2, 3 or 4) FLOW HI/LO 56C34, 56C36, 56C38, 56C40. ______

4.4.3 CCW FROM RCP SEAL HEAT EXCHANGER TEMP HI 56A09. ______

5.0 Determine the cause of the loss of CCW and correct the problem. ______

5.1 If a spurious Train A or Train B CIAS (and/or SIAS Unit 3) caused the problem, then reset the actuation system and determine the cause of the actuation. ______

5.2 If low surge tank level cause the problem, then ensure proper operation of the makeup system and inspect the CCW system for leakage. ______

COMPLETE LOSS OF THE NON-CRITICAL LOOP

*INITIALS

1.0 Attempt to restore CCW by completing this attachment within thirty (30) minutes. ________ R

1.1 If unable to restore CCW within thirty (30) minutes, then verify stopped or stop all RCPs and close HV-9218 and HV-9217, controlled bleedoff isolation valves. ________

NOTE: Controlled Bleedoff will be directed to the Quench Tank via HV-9216 and PSV-9215. N

2.0 Prepare the non-critical loop as follows:

2.1 Open all RCP seal heat exchanger return valves, TV-9144, TV-9154, TV-9164 and TV-9174. ________ O

2.2 Open CCW CNTMT isolation valves HV-6223 and HV-6236. ________

2.3 Close supply to radwaste header HV-6465. ________

2.4 Close CCW to RCPs CNTMT isolation HV-6211. ________

2.5 Override and open CCW from RCPs CNTMT isolation HV-6216. ________

3.0 If all three CCW pumps are available and two (2) CCW pumps are aligned to Train A, then override and open non-critical loop supply and return Train A valves, HV-6212 and HV-6218. ________

4.0 If all three CCW pumps are available and two (2) CCW pumps are aligned to Train B, then override and open non-critical loop supply and return Train B valves, HV-6213 and HV-6219. ________

5.0 If only two CCW pumps are available and both CCW trains are in service, then perform the following:

* The INITIAL column is an operator aid and is intended to be used as follows:

Initial each completed action. Do not write N/A. Leave blank items that are not applicable. Proceed through the instruction performing all applicable steps frequently rechecking those steps passed over to ensure action is taken when applicable.

INITIALS

5.1 Close CCW supply to both SFP heat exchangers, HCV-6545 and HCV-6546. ______

5.2 Select a CCW train to which the non-critical loop will be aligned. ______

5.2.1 Close CCW to the LTDN heat exchanger from the train selected - Train A, HV-6293B or Train B, HV-6522B. ______

5.2.2 Close CCW from the SDCHX on the train selected - Train A, HV-6500 or Train B, HV-6501. ______

5.2.3 Override and open non-critical loop supply and return on the train selected - Train A, HV-6212 and HV-6218 or Train B, HV-6213 and HV-6219. ______

6.0 Establish CCW flow to the RCPs as follows:

6.1 Open CCW supply to the reactor auxiliaries, HV-6211. ______ R

6.2 Monitor RCP seal heat exchanger outlet temperatures on TI-9144, TI-9154, TI-9164 and TI-9174. If CCW temperature from any seal heat exchanger does not stabilize below 250°F, then close the affected RCPs seal heat exchanger outlet valve and evaluate the desirability of starting the affected RCP. ______

6.3 Verify adequate flow has been restored to the RCPs as indicated by the following alarms clearing:

6.3.1 CCW FROM RCP P001(2, 3 or 4) FLOW HI/LO 56C24, 56C26, 56C28, 56C30. ______

6.3.2 CCW FROM RCP SEAL HEAT EXCHANGER TEMP HI 56A09. ______

LOSS OF CCW TO A SINGLE RCP

*INITIALS

1.0 Attempt to restore CCW to the affected RCP seal heat exchanger by completing this attachment.

1.1 If unable to restore CCW within five (5) minutes, then proceed per step 2.2.

2.0 Verify which RCP is affected due to closure of the seal heat exchanger CCW return valve:

RCP	VALVE
P-001	TV-9144
P-002	TV-9174
P-003	TV-9154
P-004	TV-9164

2.1 If RCP CCW seal heat exchanger outlet temperature is less than 300°F, indicated by TI-9144, TI-9154, TI-9164 or TI-9174, then attempt to open the affected RCP seal heat exchanger CCW return valve.

2.1.1 Monitor RT-7819, CCW process rod monitor, for indications of RCS leakage into the CCW system.

NOTE: One possible cause of high CCW seal heat exchanger outlet temperature is RCS in leakage.

2.2 If seal heat exchanger outlet temperature is greater than 300°F, as indicated by TI-9144, TI-9154, TI-9164 or TI-9174, then proceed as follows:

2.2.1 In Modes 1 and 2, trip the reactor and the turbine, and carry out Emergency Operating Instruction S023-3-5.1, "Emergency Plant Shutdown."

2.2.2 After the CEA bottom lights have been on greater than 5 seconds, stop the affected RCP.

2.2.3 Notify Instrumentation and Control to perform a circuit verification of the affected RCPs seal heat exchanger CCW return valve.

* The INITIAL column is an operator aid and is intended to be used as follows:

Initial each completed action. Do not write N/A. Leave blank items that are not applicable. Proceed through the instruction performing all applicable steps frequently checking those steps passed over to ensure action is taken when applicable.

FAILURE TO RESTORE CCW

*INITIALS

1.0 If in Modes 1 and 2 (power operation and startup), then trip the reactor and the turbine, and carry out Emergency Operating Instruction SO23-3-5.1, "Emergency Plant Shutdown." ______

1.1 After the CEA bottom lights have been on greater than 5 seconds, stop the affected RCP(s). ______

2.0 If in Modes 3, 4 or 5, then trip the affected RCP(s). ______

3.0 If all RCPs are tripped:

3.1 Close both controlled bleed-off isolation valves, HV-9218 and HV-9217. ______

NOTE: RCP Controlled Bleedoff flow will be directed to the Quench Tank via HV-9216 and PSV-9215.

3.2 Verify natural circulation flow is established per SO2(3)-3-2.31, "Natural Circulation Guidelines." ______

4.0 If available, use the Plant Monitoring System to monitor controlled bleed-off temperature on the affected RCP(s). ______

4.1 Assign applicable point ID to display CRT and line printer:

RCP/Point ID:	P001/T150	P002/T180
	P003/T160	P004/T170

______

NOTE: Normal controlled bleed-off temperatures range from 125°F - 150°F with a maximum of 180°F. Temperatures greater than 250°F are indications of seal damage. Do not start a RCP with a controlled bleed-off temperature greater than 250°F unless emergency conditions required the affected RCP(s) be started.

5.0 If CCW cannot be established, then cooldown the RCS to less than 350°F using natural circulation and place the SDC system in service and continue to cooldown to less than 250°F. ______

5.1 Since, after placing the SDC system in service, the RCPs may remain exposed to high temperature water, use the feed and bleed method to cool down the steam generators to less than 250°F. ______

* The INITIAL column is an operator aid and is intended to be used as follows:

Initial each completed action. Do not write N/A. Leave blank items that are not applicable. Proceed through the instruction performing all applicable steps frequently checking those steps passed over to ensure action is taken when applicable.

LOSS OF LOAD/REACTOR-GENERATOR MISMATCH

1.0 SYMPTOMS

The following symptoms are those which will be noted previous to a trip. The intent of this instruction is to deal with those situations leading to but not including a trip on loss of load.

1.1 Alarms

1.1.1 "Tavg./Tref. Hi"

1.1.2 "Cold Leg Temp. Hi"

1.1.3 "Turbine Bypass Demand (SBCS AWP)"

1.1.4 "CEDMCS Auto Motion Inhibit (AMI)"

1.1.5 "CEDMCS CEA Withdrawal Prohibit"

1.1.6 "Pressurizer Pressure Hi"

1.1.7 Stator Water Runback

1.1.8 Generator/Turbine Pretrip and Associated Annunciator Alarms

1.2 Indications

1.2.1 CEA's Inserting Maintaining Tavg. (CEDMCS IN "AUTO")

1.2.2 Steam Bypass Control System Auto Initiation

1.2.3 Pressurizer Sprays Auto Initiate

1.2.4 Pressurizer Heaters Initiate (Cold Water Insurge)

1.2.5 Pressurizer Level Control System Adjusts to New Level

1.2.6 Pressurizer Reliefs May Lift

1.2.7 Main Steam Reliefs May Lift

1.2.8 Generator Power Decrease

1.2.9 Turbine Govenor Valves Throttle

1.2.10 FWCS Will Reduce Flow

2.0 AUTOMATIC ACTIONS

2.1 CEDMCS Auto Insert CEA (If In Auto)

2.2 Pressurizer Sprays Lower Pressure

2.3 Pressurizer Heaters on Until 2275 psia

2.4 Pressurizer Level Control System Attempts to Maintain Program Level

2.5 FWCS Maintains S/G Levels

3.0 IMMEDIATE OPERATOR ACTIONS

3.1 If the reactor/turbine has tripped, or if the primary system is uncontrollably trending towards trip setpoints, trip the reactor/turbine and proceed to EOI "Emergency Plant Shutdown" S023-3-5.1 and end the use of this procedure.

3.2 Verify automatic actions have initiated and are maintaining the appropriate parameters as follows:

3.2.1 CEA's are inserting (If in Auto) stabilizing Tavg./Tref. mismatch.

3.2.1.1 If not, minimize Tavg./Tref. mismatch by adjusting turbine CVOL and/or inserting CEA's.

3.2.2 Pressurizer heaters cycle on until pressurizer pressure reaches 2275 psia and then turn off.

3.2.2.1 If heaters do not turn off, manually turn off.

3.2.3 Pressurizer sprays initiate to restore pressure.

3.2.3.1 If pressurizer sprays fail to initiate, manually open pressurizer spray valves and reduce pressure to 2250 psia.

3.2.4 Pressurizer level control system is maintaining program level.

3.2.4.1 If not, manually control charging and letdown to do so.

3.2.5 SBCS is operating and maintaining secondary program pressure.

3.2.5.1 If not, manually initiate SBCS and maintain secondary program pressure of 800-1000 psia dependent on power level.

3.0 IMMEDIATE OPERATOR ACTIONS (Cont'd)

3.2.5.2 If SBCS is unavailable or inadequate, use atmospheric steam dumps to maintain secondary pressure.

3.2.6 FWCS is maintaining normal steam generator levels.

CAUTION:
=======

During transient conditions, S/G levels are subject to shrink/swell syndrome; be careful not to overfeed S/G and subcool the primary.

3.2.6.1 If not, place master controller in manual and restore S/G levels.

4.0 SUBSEQUENT OPERATOR ACTION

*INITIALS

CAUTION:
=======

Do not place systems in "manual" unless misoperation in "automatic" is apparent. Systems placed in "manual"must be checked frequently to ensure proper operation.

CAUTION:
=======

Use Class IE indicators and compare indications from redundant or related indicators where possible because adverse conditions (post LOCA environment, loss of non IE instruments or instrument failure) may result in conflicting or erroneous conditions.

4.1 Manually adjust turbine load to reactor output until primary system is stable and primary to secondary balance is achieved. This shall be accomplished by:

4.1.1 The Unit operator controlling CEA position in the "manual sequential" mode and coordinating with the turbine operator. ________

4.1.2 The turbine operator shall adjust turbine load using CVOL to minimize transient conditions. ________

4.1.3 If turbine is unavailable, use SBCS to maintain primary to secondary balance. ________

4.0 SUBSEQUENT OPERATOR ACTION (Cont'd) INITIALS

*The INITIAL column is an operator aid and is intended to be used as follows:

Initial each completed action. Do not write N/A. Leave blank items that are not applicable. Proceed through the instruction performing all applicable steps frequently re-checking those steps passed over to ensure action is taken where applicable.

4.1.4 If unable to stabilize primary system and pressurizer pressure is uncontrollably trending towards trip setpoint, manually trip the reactor/turbine and commence "Emergency Plant Shutdown" S023-3-5.1 and end the use of this procedure. ________

4.2 Verify the Primary Plant System

4.2.1 Pressurizer pressure control system is stabilizing and maintaining program setpoints in the "Auto" mode. ________

4.2.1.1 If not, refer to "Pressurizer Pressure Control System Malfunction" S023-3-5.17 and perform applicable steps concurrently with the steps of this instruction. ________

4.2.2 Pressurizer Level Control System is stabilizing and maintaining program level setpoints. ________

4.2.2.1 If not, refer to EOI "Loss of Pressurizer Level Control" S023-3-5.24 and perform applicable steps concurrently with the steps of this instruction. ________

4.2.3 Feedwater Control System is restoring Steam Generator levels and is maintaining them with FWP master controller in "Auto." ________

4.2.3.1 If not, refer to "FW Regulation System Operation" S023-9-6 and perform appropriate steps concurrently with the steps in this procedure. ________

4.2.4 SBCS has closed (or is closing) turbine bypass valves. ________

4.0 SUBSEQUENT OPERATOR ACTION (Cont'd) INITIALS

4.2.4.1 If not, manually operate per "SBCS operation" S023-3-2.18 and then repeat section 4.1. ______

4.2.4.2 If Atmospheric Steam Dumps were or are in use, close dumps as turbine load is increased to pick up additional steam supply and repeat section 4.1. ______

4.3 If turbine has failed and cannot be returned to service, then

4.3.1 Commence "turbine shutdown" per S023-10-2. ______

4.3.2 Reduce reactor power to hot standby per S023-5-1.4 "Plant Shutdown from Minimum Load to Hot Standby" and perform appropriate steps concurrently with the balance of this procedure. ______

4.4 If possible, adjust boron concentration to allow full withdrawal of CEA's. ______

4.5 Determine and if possible, correct cause for transient, then

4.5.1 If unable to correct transient cause and operation is not feasible, commence "Plant Shutdown From Minimum Load to Hot Standby" per S023-5-1.4. ______

4.5.2 If below 20% F.P. and power operation is anticipated, continue power operations per S023-5-1, "Plant Startup from Hot Standby to Minimum Load." ______

4.5.3 If above 20% F.P. and power operation is anticipated, to continue, refer to "Power Operations" S023-5-1.7. ______

4.6 Notify System Operating Supervisor and inform him of the nature of the transient and current plant status. ______

H E Morgan

H. E. MORGAN
STATION OPERATIONS MANAGER

EMERGENCY PLANT SHUTDOWN

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PAGES CHANGED WITH THIS REVISION: ALL

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PROCEDURE WRITER DATE

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H. E. MORGAN, MANAGER, OPERATIONS DATE

NUS:230011

EMERGENCY PLANT SHUTDOWN

1.0 SYMPTOMS

1.1 Alarms

1.1.1 RPS Trip Path Activated

1.1.2 Reactor Trip Undervoltage Relay Tripped

1.1.3 Reactor Trip Circuit Breaker

1.1.4 PPS Channel Trouble

1.1.5 Reactor Trip/Pretrip

1.1.6 Turbine/Generator Trip/Pretrip

1.2 Indications

1.2.1 Reactor trip breakers open

1.2.2 CEA rod bottom lights on

1.2.3 Turbine control and stop valves closed

1.2.4 Moisture Separator Reheater line steam inlet valves closed

1.2.5 Unit CBs open

1.2.5.1 CB-4062 and CB-6062 for Unit 2

1.2.5.2 CB-4152 and CB-6152 for Unit 3

1.3 Key Parameters

1.3.1 Reactor power - decrease

1.3.2 Pressurizer pressure - decrease

1.3.3 RCS temperature - decrease

1.3.4 Pressurizer level - expected to decrease (until nature of emergency is known the pressurizer level should not be relied upon as an indication of RCS inventory)

1.3.5 Steam generator pressure - increase

1.3.6 Steam generator level - decrease

2.0 AUTOMATIC ACTIONS

2.1 Turbine trip (initiated by reactor tripped signal).

2.2 Unit CBs open.

2.3 Unit auxiliaries transfer to the reserve auxiliary transformers.

2.4 Steam Bypass Control System (SBCS) operates to establish and maintain steam generator pressure at 1000 psia.

2.5 Feedwater Control System reactor tripped override closes the main feedwater valves FCV-1111 & FCV-1121, positions the main feedwater bypass valves HV-1105 and HV-1106 for 5% flow (~25% open) and decreases the main feedwater pump speed to minimum.

2.6 Pressurizer Level Control System restores pressurizer level to the programmed no load setpoint (33.0% level).

2.7 Pressurizer Pressure Control System restores pressurizer pressure to ~2250 psia.

3.0 IMMEDIATE OPERATOR ACTION

CAUTION:

Do not place systems in MANUAL unless misoperation in "automatic" is apparent. Systems placed in MANUAL must be checked frequently to ensure proper operation.

CAUTION:

Use class IE indicators and compare indications from redundant or related indicators where possible because adverse conditions (post LOCA environment, loss of non-IE instruments or instrument failure) may result in conflicting or erroneous indications.

3.1 Verify all reactor trip breakers indicate open and reactor power is decreasing.

3.1.1 If the reactor is not tripped, then push all four manual reactor trip pushbuttons.

3.0 IMMEDIATE OPERATOR ACTION (Continued)

3.1.2 If ten or more CEAs are not fully inserted into the core, or if the reactor excore linear power is not below 6%, then perform the following steps simultaneously (ATWS actions):

3.1.2.1 De-energize Load Centers B15 and B16.

3.1.2.2 Manually initiate EFAS #1 and EFAS #2.

3.1.2.3 Initiate Emergency Boration.

3.1.3 If more than one CEA has not fully inserted, then commence Emergency Boration.

3.2 Verify the turbine tripped and all HP Stop and HP Governor valves are closed.

3.2.1 If the turbine is not tripped, then push the manual trip pushbutton.

3.3 When the "Generator Protection Trip" alarm is received, then verify that the unit auxiliaries have transferred to the reserve auxiliary transformers.

3.3.1 If the unit auxiliaries have not transferred, then manually transfer them to the reserve auxiliary transformers.

3.4 When the auxiliaries have transferred, then verify that the unit output breakers are open and turbine speed is decreasing.

3.4.1 If the unit output breakers are not open, then manually open the unit output breakers.

3.5 Verify that the following key parameters are trending normally:

3.5.1 Steam Generator Pressure

3.5.2 RCS Tavg

3.5.3 Pressurizer Pressure

3.5.4 Pressurizer Level

3.5.5 Total Feedwater Flow (MFW and/or AFW)

3.5.6 Steam Generator Level

3.0 IMMEDIATE OPERATOR ACTION (Continued)

3.6 If key parameters are not following normal trends, then place system in MANUAL and restore to normal.

3.7 Use the Public Address System to notify on-site personnel concerning the nature of the emergency.

4.0 SUBSEQUENT OPERATOR ACTION

INITIALS*

4.1 Verify all immediate operator actions have been initiated as follows:

4.1.1 Verify all reactor trip breakers indicate open and reactor power is decreasing. ________

4.1.1.1 If the reactor has not tripped, then push all four manual reactor trip pushbuttons. ________

4.1.2 If 10 or more CEAs have not fully inserted into the core, or if the reactor excore linear power is not below 6%, then verify completed or complete the following:

4.1.2.1 De-energize Load Centers B15 and B16. ________

4.1.2.2 Manually initiate EFAS #1 and EFAS #2. ________

4.1.2.3 Start all available charging pumps and commence emergency boration. ________

4.1.2.4 Complete followup ATWS actions per Attachment 3 of this instruction. ________

4.1.3 If more than one CEA has not fully inserted, then commence emergency boration per SO23-3-5.10, "Emergency Boration of Reactor Coolant System," performing applicable steps concurrently with the steps in this instruction. ________

*The INITIAL column is an operator aid and is intended to be used as follows:

Initial each completed action. Do not write N/A. Leave blank items that are not applicable. Proceed through the instruction performing all applicable steps frequently re-checking those steps passed over to ensure action is taken where applicable.

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.4 Verify the turbine is tripped and all HP Stop and HP Governor valves are closed. ______

4.1.4.1 If the turbine is not tripped, then manually trip the turbine. ______

4.1.4.2 If any HP Stop Valve and its associated in-line HP Governor Valve remains open, then close the MSIVs. ______

4.1.4.3 If unable to trip the turbine from the control room, then dispatch an operator to open the DC supply breaker to the Unitized Actuator dump valve solenoids (D5P489). ______

4.1.4.4 Complete followup Turbine Trip Actions per Attachment 7. ______

4.1.5 When the "Generator Protection Trip" alarm is received or one minute after the turbine is tripped, then verify open or open the Unit Auxiliary Transformer low side breakers, and then verify open or open the unit output breakers. ______

4.1.5.1 If, after the unit output breakers are open and turbine speed increases to 2000 rpm, then close the MSIVs. ______

4.1.5.2 If the Generator Excitation system does not function to reduce and suppress the 22KV voltage, then dispatch an operator to open the AVR AC supply breakers 29VA and 29VB. ______

4.1.6 Verify energized or energize the following 6.9KV and 4KV Buses and check RCPs running: ______

4.1.6.1 If both A01 and A02 cannot be energized, then use S02(3)-3-2.31, Natural Circulation Guidelines, to confirm that natural circulation has been established, performing applicable steps concurrently with the steps in this instruction. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.6.2 If both A03 and A07 cannot be energized, then use the atmospheric steam dumps and the auxiliary feedwater system to control steam generator pressure and level. ______

4.1.6.3 If A08 and A09 cannot be energized, then carry out S023-3-5.38, "Loss of Non-IE Instrumentation Buses," performing applicable steps concurrently with the steps in this instruction. ______

4.1.6.4 If both A04 and A06 cannot be energized, then carry out S023-3-5.4, "Complete Loss of Offsite Electrical Power," performing applicable steps concurrently with the steps in this instruction. ______

4.1.6.5 If offsite power is available to the opposite unit and if A01, A02, A03, A07, A08 or A09 buses cannot be energized, then carry out S023-3-5.4.1, "Loss of Offsite Power to a Unit," performing applicable steps concurrently with the steps in this instruction. ______

4.1.6.6 If all offsite AC power is lost, then carry out S023-3-5.4, "Complete Loss of Offsite Electrical Power," performing applicable steps concurrently with the steps in this instruction. ______

4.1.7 Verify that the following key parameters are trending normally toward their Hot Standby condition. ______

4.1.7.1 Steam Generator (SG) pressure approaching 1000 psia (between 950 psia - 1050 psia). ______

4.1.7.1.1 If SG pressure is not in range, then place SBCS valve permissive in MANUAL per S023-3-2.18, "Steam Bypass System Operation," and attempt to restore SG pressure to ~ 1000 psia. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.7.1.2 If SG pressure remains above 1050 psia, then operate the Atmospheric Dump Valves per SO23-3-2.18.1, "Atmospheric Steam Dump System Operation," to reduce SG pressure and close and/or isolate the SBCS system. ________

4.1.7.1.3 If SG pressure remains below 950 psia, then complete the following:

4.1.7.1.3.1 Verify SBCS valves are closed or place SBCS in MANUAL per SO23-3-2.18 and close SBCS valves. ________

4.1.7.1.3.2 Verify Turbine Stop Valves are closed. ________

4.1.7.1.3.3 Verify Atmospheric Dump Valves are closed or place system in MANUAL per SO23-3-2.18.1 and close Atmospheric Dump Valves. ________

4.1.7.1.3.4 If Auxiliary Feedwater is available, then stop Main Feedwater Pumps per SO23-2-1, "Main Feedwater Pump and Turbine Operation," and feed SG per SO23-2-4, "Auxiliary Feedwater System Operation." ________

4.1.7.1.3.5 Verify Live Steam to Reheaters closed. ________

4.1.7.1.4 If SG Pressure is below 850 psia, then close the MSIVs. ________

4.1.7.1.4.1 If Auxiliary steam is anticipated to be lost, or is lost, then align Auxiliary Steam to the non-affected Unit if available, or Start the Auxiliary Boiler to supply Auxiliary Steam to the affected Unit. ________

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.7.2 RCS Tavg is approaching 545°F (between 550°F - 540°F). ______

4.1.7.2.1 If RCS Tavg is not in range, then adjust Steam Generator Pressure to control Tavg and/or adjust feedwater flow rate to control Tavg. ______

4.1.7.2.1.1 Refer to Attachment 9, "Actions for Excessive Cooldown." ______

4.1.7.3 Pressurizer pressure approaching 2250 psia (between 2025 psia - 2275 psia). ______

4.1.4.3.1 If Pressurizer Pressure is above 2275 psia, then complete the following:

4.1.7.3.1.1 Restore RCS Tavg to less than 550°F per step 4.1.7.2.1. ______

4.1.1.3.1.2 Verify proper PZR Spray Operation or place PZR Pressure Control in MANUAL per S023-3-1.10, "Manual Pressurizer Pressure Control," and reduce pressure. ______

4.1.7.3.1.2.1 If normal PZR Spray is unavailable, then initiate Auxiliary Spray per S023-3-5.17, "Pressurizer Pressure Control Malfunction." ______

4.1.7.3.1.3 Verify that PZR Heaters are deenergized or turn Heaters OFF. ______

4.1.7.3.2 If PZR Pressure is below 2025 psia, then complete the following:

4.1.7.3.2.1 Restore RCS Tavg to greater than 540°F. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.7.3.2.2 Verify Proper PZR Spray valve operation or place PZR Pressure Control in MANUAL and close spray valves per S023-3-1.10. ______

4.1.7.3.2.3 Turn ON all available PZR Heaters. ______

4.1.7.3.2.4 Immediately complete Attachment 8, "Verification of RCS Pressure Boundary," of this instruction. ______

4.1.7.4 Pressurizer Level approaches 33% (between 15% - 42%). ______

4.1.7.4.1 If PZR level is not in range, then complete the following:

4.1.7.4.1.1 Restore RCS Tavg to between 540°F to 550°F. ______

4.1.7.4.1.2 Verify RCS subcooled at least 50°F by using Subcooling Margin Monitor on Unit 2 and the QSPDS/CFMS on Unit 3. ______

4.1.7.4.1.3 Adjust Charging flow and Letdown flow per S023-3-5.24, "Loss of Pressurizer Level Control," to restore level to between 33% to 42%. ______

4.1.7.5 Total Feedwater flowrate controlling at ~5% (between 3% - 7%). ______

4.1.7.5.1 Close Main Feedwater Regulator Block Valves HV-4051 and HV-4047. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued) — INITIALS

4.1.7.5.2 If total feedwater flow is not in range, then use Main Feedwater System Controls in conjunction with Auxiliary Feedwater System controls in MANUAL to restore feedwater flow to ~5% per S023-2-1, "Main Feedwater Pump and Turbine Operation," S023-9-6, "Feedwater Regulating System Operation," and S023-2-4, "Auxiliary Feedwater System Operation." ________

NOTE:

Use the following conversions to approximate 3% feedwater flow on FR(s) 1011 and 1021:

On Blue Pen ~ .75 which is ~ 225,000 lbs/hr
On Violet Pen ~ 1.4 which is ~ 450 gpm

4.1.7.6 Steam Generator (SG) level approaches 67% Narrow Range (NR) (between 50%NR to 80%NR). ________

> **CAUTION:**
>
> Overfilling steam generators causes a rapid RCS cooldown transient and SIAS actuation on low pressure, and can cause water hammer in the Main Steam System and large quantities of subcooled water to collect in steam line. These conditions can result in damaged pipe supports, potential steamline breaks or damage to valves in the Main Steam System.

4.1.7.6.1 If SG level is above 80%NR, then stop all Main and Auxiliary Feedwater flow to the affected SG(s). ________

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.7.6.2 If level in either steam generator goes offscale high as indicated by 2 or more channels, initiate MSIS and perform the Steam Generator Overfill followup actions per Attachment 4 of this instruction. ______

4.1.7.6.3 If SG level is less than 50%NR, then maintain level constant unless RCS Tavg is above 540°F, then gradually restore level to ~67%NR. ______

4.1.7.7 If Steam Generator Pressure and level cannot be restored, then complete the following:

4.1.7.7.1 Verify bypass regulator flow on FR-1011 and FR-1021. ______

4.1.7.7.2 If Main Feedwater Regulator Bypass flow is less than 3% and EFAS #1 and EFAS #2 have not actuated automatically, then manually actuate EFAS #1 and EFAS #2 by depressing each actuation pushbutton only once. ______

CAUTION:

Each EFAS #1 and EFAS #2 pushbutton must be depressed to insure full actuation.

NOTE:

To reset the manual actuation signal, depress the actuation pushbuttons again.

NOTE:

EFAS Actuation trips the Train A & B pressurizer backup heaters OFF. An override is provided so heaters can be used during an EFAS Actuation.

4.0 SUBSEQUENT OPERATOR ACTIONS (Continued) INITIALS

4.1.7.7.3 If EFAS #1 is actuated, then check flow on FI-4725. If flow is less than 700 gpm, then verify started or start auxiliary feedwater pumps P-141 and P-140 and auxiliary feedwater control valves HV-4706 and HV-4713 and isolation valves HV-4715 and HV-4731. ______

4.1.7.7.4 If EFAS #2 is actuated, check flow on FI-4720. If flow is less than 700 gpm, then verify started or start auxiliary feedwater pumps P-504 and P-140 and open auxiliary feedwater control valves HV-4705 and HV-4712, and isolation valves HV-4714 and HV-4730. ______

4.1.7.7.5 After EFAS #1 or EFAS #2 manual actuation and when steam generator narrow range level increases to 30% manually control steam generator levels by manually depressing each actuation pushbutton once again. ______

4.1.8 If at least 350 gpm feedwater flow cannot be established to each steam generator, then perform the following:

4.1.8.1 If steam generator pressure has rapidly decreased to the MSIS setpoint, then go to S023-3-5.9, Steam Line Rupture, performing applicable steps concurrently with the steps in this instruction. ______

4.1.8.2 If steam generator pressure shows a gradual pressure change (increase or decrease, then go to S023-3-5.30, Loss of Feedwater, performing applicable applicable steps concurrently with the steps in this instruction. ______

4.1.8.3 If unable to establish feedwater flow to at least one steam generator, then establish emergency feedwater flow by performing the following:

4.1.8.3.1 Verify aligned or align auxiliary feedwater pump's manual valving. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.1.8.3.2 Vent auxiliary feedwater piping and pump casings. ______

4.1.8.3.3 Start an auxiliary feedwater pump. ______

4.1.9 Use the Public Address System or other means of communication and perform the following: (If required, then Override BS-11 and BS-12, EFS signals, at MCC BS).

4.1.9.1 Announce Reactor Trip Unit 2(3). ______

4.1.9.2 Call the Shift Supervisor to the Control Room. ______

4.1.9.3 Call the Shift Technical Advisor to the Control Room. ______

4.1.9.4 Call the Nuclear Operations Assistant and the Outside Operators to the Control Room. ______

4.1.9.5 Clear the Control Room of all non-essential personnel. ______

4.1.9.6 Designate and announce by name the SRO in charge. Direct all status reports and requests for approvals to take corrective actions to the SRO "in charge." ______

4.2 The Shift Supervisor shall notify the Plant Superintendent or designee and Shift Technical Advisor and discuss the situation. ______

4.2.1 An assessment of the plant status and safety shall be made and the event classified per S023-VIII-11, "Recognition and Classification of Emergencies." ______

4.2.1.1 If available, then use the PAMI wide range trend recorders, PMS alarm history, and POST Trip Review printouts, CFMS history tape outputs, CPC and CEAC buffer outputs, and the Turbine Trip Flags to aid in determining the cause of the trip and in assessing trends in plant status. ______

4.0 <u>SUBSEQUENT OPERATOR ACTION</u> (Continued) INITIALS

4.2.2 <u>If</u> an emergency is declared (Unusual, Alert, Site Emergency or General Emergency), <u>then</u> use the following Emergency Procedures to implement the SONGS 1, 2 and 3 Emergency Plan:

4.2.2.1 Unusual - S0123-VIII-12 ________

4.2.2.2 Alert - S0123-VIII-13 ________

4.2.2.3 Site Emergency - S0123-VIII-14 ________

4.2.2.4 General Emergency - S0123-VIII-15 ________

4.2.3 Notify the NRC via the Red Phone within one hour, per S023-0-5. ________

4.2.3.1 If possible, discuss the contents of the notification with the Plant Superintendent or designee prior to the notification. ________

4.2.3.2 Record the names of the persons involved with the notification and time:

____
SCE

________________________ ________ ________
NRC Time

4.2.4 Notify the Systems Operating Supervisor concerning the nature of the emergency. ________

4.3 Determine if the RCS pressure boundary is intact by completing Attachment 8. ________

4.3.1 <u>If</u> determined by Attachment 8 that a Pressurizer Safety Valve has actuated, <u>then</u> ensure that notification to NSSS Engineering of the event is made so that arrangements for a VT-3 inspection can be made. ________

4.4 Use S023-3-2.30, "Determination of Adequate Core Cooling," to confirm that conditions are not trending toward an inadequate core cooling event, performing applicable steps concurrently with the steps in this instruction. ________

4.0 SUBSEQUENT OPERATOR ACTION (Continued) — INITIALS

4.5 *If* main feedwater is available, *then* verify proper operation of the Reactor Tripped/Feedwater Control System Override by completing the following:

> **CAUTION:**
>
> *If* both main and auxiliary feedwater are supplying the steam generators, *then* rapid action (i.e., within 2 minutes) *may* be required to avoid an excessive feedwater addition cooldown transient. (Tavg below 540°F and decreasing).

4.5.1 Ensure closed the Main Feedwater Block Valves, HV-4051 and HV-4047. ________

4.5.2 Place HIC-1111 and HIC-1121 in MANUAL/CLOSED as follows:

4.5.2.1 Match the Manual Output Indicator to the Auto Output Indicator. ________

4.5.2.2 Place AUTO/MANUAL X-fer Sw. in MANUAL. ________

4.5.2.3 Adjust Manual Control knob as required to decrease manual output to "zero." ________

4.5.3 Verify Main Feedwater Bypass valves, HV-1105 and HV-1106, throttled to the 5% flow position or are in the process of closing to the 5% flow position. (HIC-1105 and HIC-1106 indicate ~70% or less demand, and valve position indication is at ~70% or less). ________

4.5.4 *If* Steam Generator(s) level is increasing greater than 2% per minute (narrow range indication) with Tavg <545°F, place HIC-1105 and/or HIC-1106 to the affected Stm. Gen. in MANUAL and throttle feed flow as follows:

4.5.4.1 Match the Manual Output Indicator to the Auto Output Indicator. ________

4.0 <u>SUBSEQUENT OPERATOR ACTION</u> (Continued) INITIALS

4.5.4.2 Place AUTO/MANUAL X-fer Sw. in MANUAL. ______

4.5.4.3 Adjust Manual Control knob as required to reduce feedwater flow to affected Stm. Gen.(s). ______

4.5.5 Verify that main feedwater pump's speed is decreasing to or at ~3200 rpm as indicated on SR-4500 and SR-4501. ______

4.5.6 <u>If</u> either main feedwater pump does not indicate speed is decreasing to or at ~3200 rpm, <u>then</u> complete the following:

4.5.6.1 On HIC-1108 (FW Pump Speed Setpoint K006) and/or HIC-1107 (FW Pump Speed Setpoint K005) use the manual control knob to match the manual output indicator with the auto output indicator. ______

4.5.6.2 Transfer the auto/manual transfer switch to manual. ______

4.5.6.3 Using the manual control knob set the affected main feedwater pump(s) speed to ~3200 rpm as indicated on SR-4500 (K-006) and/or SR-4051 (K-005). ______

<u>CAUTION:</u>

Do not reduce feedwater flow below the requirement to maintain a constant or increasing level.

4.6 <u>If</u> there is an undesirable cooldown of the RCS (Tavg below 540°F and decreasing), <u>then</u> complete Attachment 9. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued)

INITIALS

4.7 After steam generator narrow range levels reach 30%, reset EFAS-1 and EFAS-2 at the manual actuation station, if required, and at the Train A and Train B ESFAS Auxiliary Relay Cabinets. ________

NOTE:

Resetting EFAS-1 and EFAS-2 disables the override-stop signals, realigning EFAS for automatic actuation of the auxiliary feedwater pumps and valves. (Failure to do so could result in a loss of heat sink.)

4.7.1 When EFAS-1 is reset, open HV-4731. When HV-4731 is open, verify closed or close HV-4715. ________

4.7.1.1 If HV-4731 cannot be opened, then open HV-4715. ________

4.7.2 When EFAS-2 is reset, open HV-4714. When HV-4714 is open, verify closed or close HV-4730. ________

4.7.2.1 If HV-4714 cannot be opened, then open HV-4730. ________

NOTE:

HV-4730 and HV-4715 should remain closed except when an EFAS is present or during surveillance testing. This avoids a steam line break scenario concurrent with the loss of 125 VDC buses D1 or D2 which would result in feeding a ruptured steam generator.

4.7.3 If the steam generator blowdown is necessary for chemistry control, or to facilitate steam generator level reduction, then establish steam generator blowdown per S023-9-4, "Steam Generator Blowdown Processing System Operation. ________

NOTE:

EFAS isolates Steam Generator Blowdown.

4.7.4 Stop the Auxiliary Feed Pump Room IE HVAC units A-394 and A-443. ________

4.0 SUBSEQUENT OPERATOR ACTION (Continued) — INITIALS

4.8 Establish auxiliary feedwater flow to each steam generator and secure the main feedwater flow as follows:

4.8.1 Verify started or start auxiliary feedwater pump P-141. ________

4.8.1.1 Jog open or close as necessary HV-4713 by intermittently depressing HS-4713-1 to establish a gradual (less than 2% per minute) S/G#1 level recovery to ~65% narrow range level. ________

4.8.1.2 If P-141 cannot be started or if HV-4713 cannot be positioned as desired, verify started or start P-140. ________

4.8.1.2.1 Jog open or closed as necessary HV-4706 by intermittently depressing HS-4706-2 to establish a gradual S/G#1 level recovery to ~65% narrow range level. ________

4.8.1.3 Close main feedwater bypass valve HV-1105 using HIC-1105. ________

4.8.2 Verify started or start auxiliary feedwater pump P-504. ________

4.8.2.1 Jog open or closed as necessary HV-4712 by intermittently depressing HS-4712-1 to establish a gradual (less than 2% per minute) S/G#2 level recovery to ~65% narrow range level. ________

4.8.2.2 If P-504 cannot be started or if HV-4712 cannot be positioned as desired, verify started or start P-140. ________

4.8.2.2.1 Jog open or closed as necessary HV-4705 by intermittently depressing HS-4705-2 to establish a gradual S/G#2 level recovery to ~65% narrow range level. ________

4.8.2.3 Close main feedwater bypass valve HV-1106 using HIC-1106. ________

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.8.3 Stop main feedwater pumps per S023-2-1, "Main Feedwater Pump and Turbine Operation." ______

4.8.4 If Condensate Storage Tank T-121 level is decreasing, then establish makeup to the tank per Attachment 5. ______

4.8.5 If auxiliary feedwater flow required is below the minimum limits for the normal control valves, then control flow with the two inch bypass valves.

CAUTION:

To prevent damage to valve seats, do not throttle valves HV-4705, 4706, 4712, and 4713 less than 200 gpm (10% open) at normal steam generator pressure and 520 gpm (26% open) at cold steam generator conditions.

NOTE:

Local SG level and feed flow indications may be used as operator aid, however indications must be verified by Control Room Indication which is the only official indication and shall be used for establishing any limits as the controlling indication.

CAUTION:

Misalignment of the 2" crosstie can result in backward flow through "kerotest" type valves which will result in valve damage and potential excessive vibration. Strict compliance with the following lineup is mandatory.

4.8.5.1 Station an operator in the auxiliary feedwater pump rooms during the entire time the two inch bypass valves are being used. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.8.5.2 Communications must be established between the control room and the auxiliary feedwater pump room and shall be maintained throughout the operation. ________

4.8.5.3 If both aux. feedwater pumps are to be used, then verify P-504 and P-141 running and open the following:

S2(3)1305MU553 ________

S2(3)1305MU152 ________

S2(3)1305MU154 ________

S2(3)1305MU153 ________

4.8.5.4 If P-504 is to be used to feed both steam generators, then verify P-504 running and open the following:

S2(3)1305MU553 ________

S2(3)1305MU152 ________

S2(3)1305MU153 ________

4.8.5.5 If P-141 is to be used to feed both steam generators, then verify P-141 running and open the following:

S2(3)1305MU154 ________

S2(3)1305MU153 ________

S2(3)1305MU152 ________

4.8.5.6 Under the direction of the control room operator, throttle open S2(3)1305MU549 (HV-4712 Bypass) while the control room operator closes HV-4712 maintaining level in steam generator E-088. ________

4.8.5.7 Under the direction of the control room operator, throttle open S2(3)1305MU551 (HV-4713 Bypass) while the control room operator closes HV-4713 maintaining level in steam generator E-089. ________

4.0 SUBSEQUENT OPERATOR ACTION (Continued) INITIALS

4.8.5.8 Make adjustments as necessary to S2(3)1305MU549 and S2(3)1305MU551 to maintain steam generator level. ______

4.8.5.9 If it is desired to stop an aux. feed pump while two aux. feed pumps are running, then close the pumps associated bypass supply (P-504 S2(3)1305MU553, P-141 S2(3)1305MU154) prior to securing the pump. ______

4.8.5.10 If desired to secure all feedwater, then close all bypass valves (supply, control, outlet) and then secure aux. feedwater pumps. ______

4.9 Notify the Chemistry Dept. to perform sampling of the gaseous release paths identified in Tech. Spec. 4.11.2.1.2, Table 4.11-2. ______

4.10 Within thirty (30) minutes after a reactor trip, place both startup channel alarms in service per S023-3-2.15, "Excore Instrumentation Operation." ______

4.10.1 If both startup channels are not operating, then perform the Dilution Accident Detection actions per Attachment 6. ______

4.11 Verify a shutdown margin greater than 5.15% Δ K/K per S023-3-3.29, "Determination of Shutdown Margin." ______

4.11.1 If shutdown margin is less than 5.15% Δ K/K, then immediately start emergency boration per S023-3-5.10, "Emergency Boration of Reactor Coolant System," performing applicable steps concurrently with the steps in this instruction. ______

4.12 If an extended shutdown is anticipated, then continue operations per S023-5-1.5, "Plant Shutdown from Hot Standby to Cold," and terminate the use of this instruction. ______

4.13 If a plant startup is anticipated, then continue operations per S023-5-1.3, "Plant Startup from Hot Standby to Minimum Load," and terminate the use of this instruction. ______

4.0 SUBSEQUENT OPERATOR ACTION (Continued)

INITIALS

4.14 If the reactor tripped from 100% power, then complete the "Reactor Trip Cycles," Attachment of S023-0-20, "Cumulative Equipment Inoperability and Design Cycles. ______

4.15 If the reactor tripped from 15% power or greater, then initiate an iodine sample within 2 to 6 hours after trip. ______

4.16 Complete records as required for review and file per S023-0-28, "Operating Records," and S023-0-11, "Startup and Shutdown Chart Removal and Identification." ______

5.0 ATTACHMENTS

5.1 Attachment 1 - Small Break Accident Identification (1 page)

5.2 Attachment 2 - Large Break Accident Identification (1 page)

5.3 Attachment 3 - Anticipated Transient Without Scram Followup Actions (1 page)

5.4 Attachment 4 - Steam Generator Overfill Followup Actions (2 pages)

5.5 Attachment 5 - Condensate Tank T-121 Makeup Actions (2 pages)

5.6 Attachment 6 - Dilution Accident Detection Actions (1 page)

5.7 Attachment 7 - Actions on Turbine Trip (2 pages)

5.8 Attachment 8 - Verification of RCS Pressure Boundary (2 pages)

5.9 Attachment 9 - Actions for Excessive Cooldown (2 pages)

6.0 REFERENCES

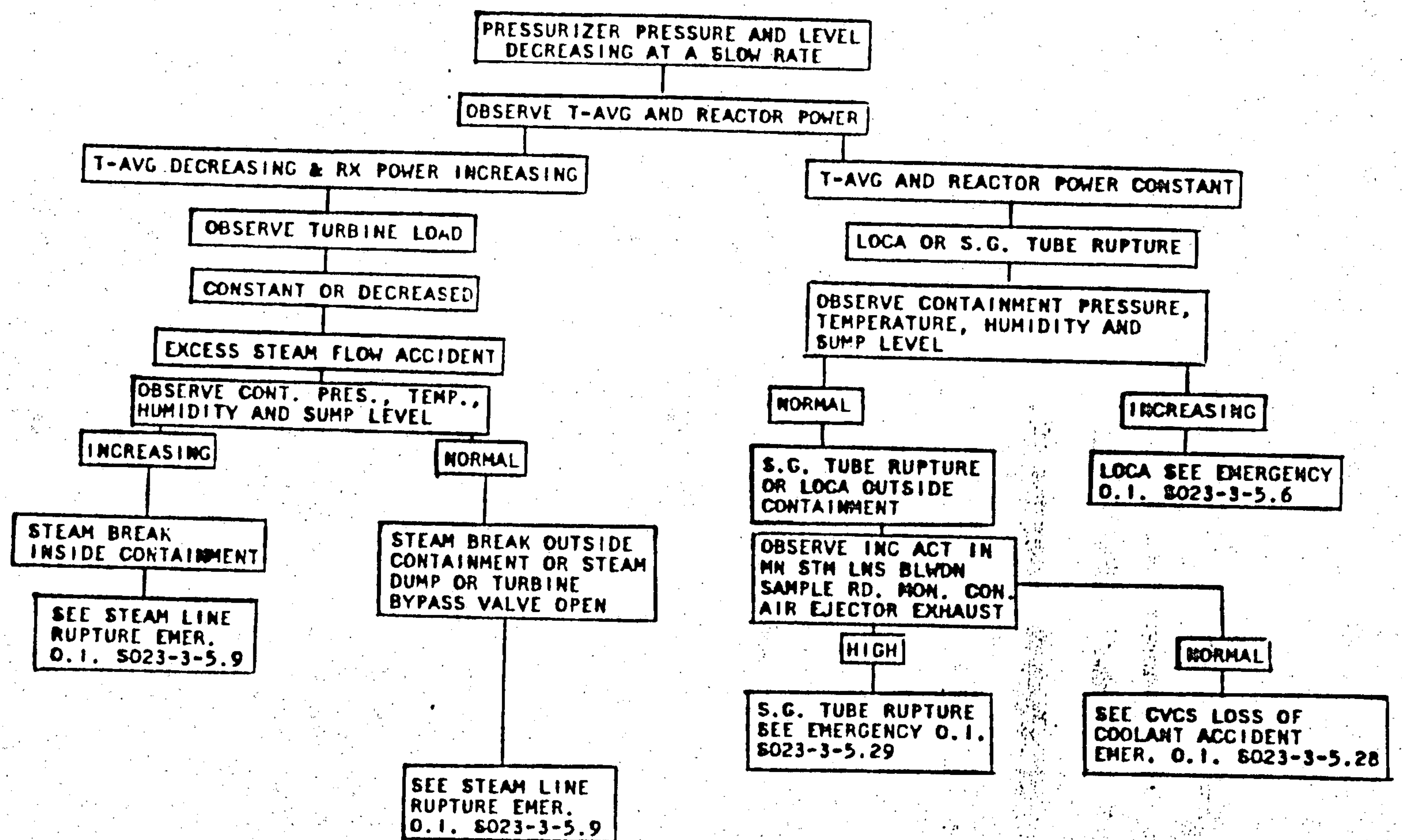
6.1 None

7.0 RECORDS

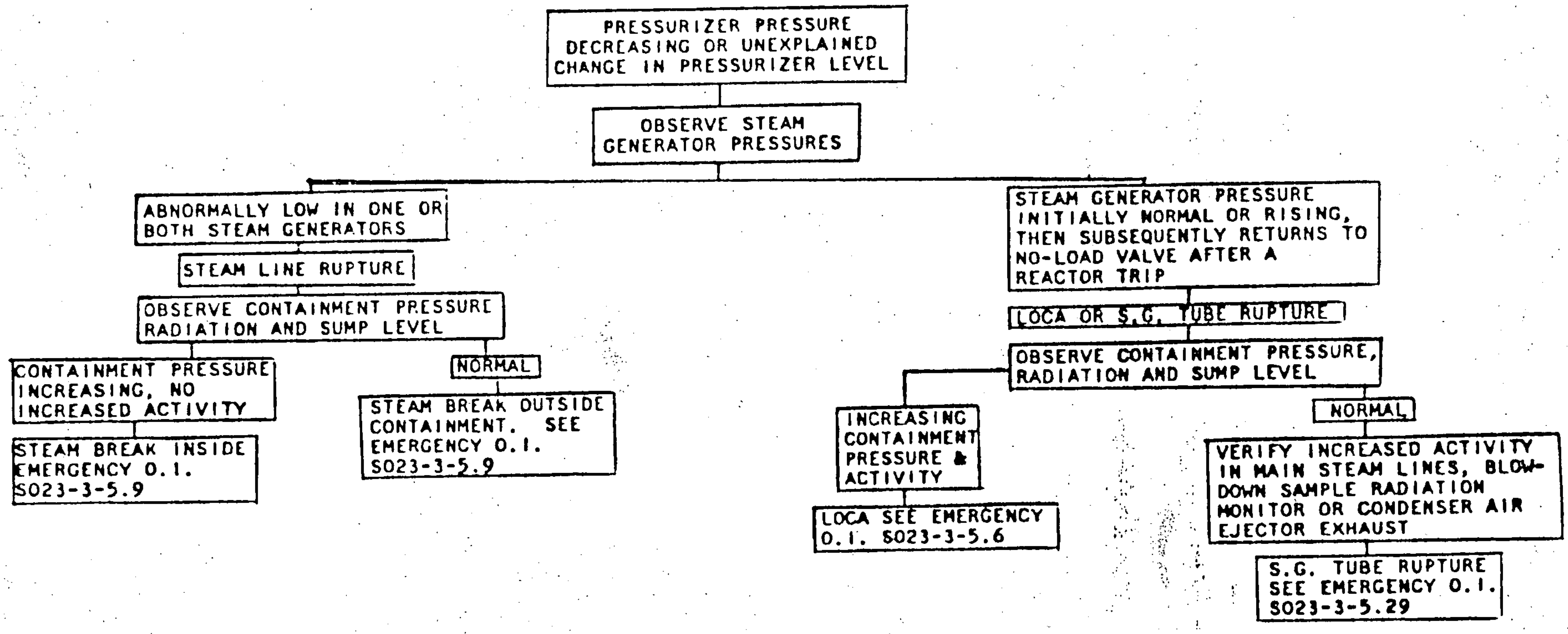
7.1 Upon termination of this procedure and applicable attachments, file in the Operations Evolutions File.

7.2 Completed procedure and attachments shall be transmitted to CDM for retention and storage, in accordance with applicable station procedures.

ACCIDENT IDENTIFICATION SMALL BREAK



ACCIDENT IDENTIFICATION LARGE BREAK



ANTICIPATED TRANSIENT WITHOUT SCRAM FOLLOWUP ACTIONS

INITIALS

1.1 Upon receipt of the "CEDMCS Bus Undervoltage" annunciator, re-energize 480 load center B16. ________

1.2 Dispatch an operator to open the reactor trip breakers locally and open both M/G sets input and output breakers, per S023-3-2.19.1, "CEDM MG Set Operation." ________

1.2.1 When the input and output breakers for both M/G sets have been opened, then re-energize 480V load center B15. ________

1.3 Verify adequate feedwater flow for the existing reactor power. ________

1.3.1 If auxiliary feedwater flow to S/G#1 is less than 700 gpm (FI-4725), then verify started or start auxiliary feedwater pumps P-141 and P-140, open auxiliary feedwater control valves HV-4706 and HV-4713 and isolation valves HV-4715 and HV-4731. ________

1.3.2 If auxiliary feedwater flow to S/G#2 is less than 700 gpm (FI-4720), then verify started or start auxiliary feedwater pumps P-504 and P-140, open auxiliary feedwater control valves HV-4705 and HV-4712 and isolation valves HV-4714 and HV-4730. ________

1.3.3 If the Reactor Tripped/Feedwater Override has functioned to reduce main feedwater flow to less than the existing reactor power, then take manual control of the main feedwater valves and pumps and increase main feedwater flow to match reactor power. ________

1.4 Verify maximum emergency boration rate. ________

1.4.1 Place all three charging pumps in Manual and start available charging pumps. ________

1.4.2 Verify the emergency boration flow path is established per S023-3-5.10, "Emergency Boration," performing applicable steps concurrently with the steps in this instruction. ________

1.5 Continue to match feedwater flow to reactor power until a reactor shutdown margin of at least 7.25% $\Delta K/K$ has been established. ________

1.6 When the ATWS followup actions have been initiated, then continue to perform the subsequent actions of this instruction. ________

1.7 When the ATWS followup actions have been completed, then notify the Shift Supervisor and terminate use of this attachment. ________

STEAM GENERATOR OVERFILL FOLLOWUP ACTIONS

INITIALS

1.1 Verify that Train A and Train B MSIS actuation alarms are received. ________

1.1.1 If both Train A and Train B MSIS actuation alarms are not received, dispatch an operator to open the associated trip leg circuit breakers at Auxiliary Relay Cabinets A and B. ________

1.2 After MSIS is actuated, verify closed or close the following:

1.2.1 HV-8205 and HV-8204, Main Steam Isolation Valves. ________

1.2.2 HV-8203 and HV-8202, Main Steam Isolation Bypass Valves. ________

1.2.3 HV-8200 and HV-8201, Auxiliary Feedwater Pump Isolation Valves. ________

1.2.4 HV-8419 and HV-8421, Atmospheric Steam Dump Valves. ________

1.2.5 HV-4048 and HV-4052, Main Feedwater Isolation Valves. ________

1.2.6 HV-4715 and HV-4731, Auxiliary Feedwater to S/G#1 Isolation Valves. ________

1.2.7 HV-4714 and HV-4730, Auxiliary Feedwater to S/G#2 Isolation Valves. ________

1.3 Reset MSIS per S023-3-2.10, Main Steam Isolation Valve Operation, and open the following valves:

1.3.1 HV-4053 and HV-4057, Blowdown and Sample Isolation Valves. ________

1.3.2 HV-4054 and HV-4058, Blowdown and Sample Isolation Valves. ________

1.3.3 HV-8249 and HV-8248, Main Steam Drain Isolation Valves. ________

1.3.4 Use the blowdown processing system to restore steam generator narrow range level to ~65%. ________

1.4 Use all available traps to drain the steam leads to minimize the potential for water hammer per S023-2-9, "Placing the Main Steam Leads in Service," performing applicable steps concurrently with the steps in this instruction. ________

INITIALS

1.5 Inspect the main steam leads for any indication of structural or pipe support damage. ______

1.5.1 Any structural or pipe support damage which renders the main steam leads unavailable for service is to be referred to the On-Site Review Committee for evaluation. ______

1.6 When the steam generator levels are reduced to less than 90% and the lines upstream and downstream of the MSIVs have been drained, and if no structural or pipe support damage exists, then reopen the steam generator isolation valves listed in Step 1.2 of this attachment. ______

1.7 When the Steam Generator Overfill followup actions have been initiated, then continue to perform the subsequent actions of this instruction. ______

1.8 When the Steam Generator Overfill actions have been completed, then notify the Shift Supervisor and terminate use of this attachment. ______

CONDENSATE TANK T-121 MAKEUP ACTIONS

INITIALS

1.0 MAKEUP ACTIONS

1.1 Provide makeup to condensate tank T-121 as follows: (Listed in order of preference).

1.1.1 If available, place the Makeup Demineralizer in service per S023-11-2, Makeup Demineralizer Operation, and lineup to Condensate Storage Tank T-121. ______

1.1.2 If the makeup demineralizer is not available, line up condensate tank T-120 to gravity feed to T-121. Shut condensate tank T-120 makeup and draw off isolation S2(3)1414MU092. Open or check open condensate tanks T-120 and T-121 cross connect S2(3)1305MU476 and S2(3)1414MU052. ______

1.1.3 Cross tie to other unit's T-120 per S023-9-5, Condensate Transfer and Storage. ______

1.2 If T-121 level decreases to 70% and no source other than Fire Main water is available for makeup, initiate a rapid plant cooldown and place the Shutdown Cooling System in service. ______

1.3 If T-121 makeup is required and no sources are available per Step 1.1, line up Fire Protection System water by:

1.3.1 Verifying the Diesel Driven Fire Pump is running. ______

.1 "Fire Pump P-220(E) Running" alarm on annuciator panel 2/3-UA-61A window 11 in the control room will be annunciated. ______

.2 If the Diesel Driven Fire Pump is not running, then start from the control room. ______

1.3.2 Closing locked open valve S2(3)1305MU082, condensate tank cross-connect line drain valve. ______

1.3.3 Opening locked closed valve S2(3)1305MU474, fire water supply to T-121. ______

NOTE: S2(3)1305MU474 has a unique lock with the key located in the Shift Supervisor's office.

INITIALS

1.4 If Condensate Storage Tank T-120 has ruptured, and T-121 level has decreased below the water level contained in the T-120 vault, then enter the T-121 vault and open the T-120 vault to T-121 emergency supply valve S2(3)1305MU088. ______

1.5 When the T-121 makeup actions have been completed, then notify the Shift Supervisor and terminate use of this attachment. ______

DILUTION ACCIDENT DETECTION ACTIONS

INITIALS

1.1 If both startup channels are not operating when the Excore NIS Log Channels decrease below 10^{-6} % power, perform the following:

1.1.1 Notify the I&C section and request repair or replacement of the affected startup channels. ______

1.1.2 Immediately, and then at intervals as indicated on Table 1 below, perform the Dilution Accident Monitoring actions as follows:

1.1.2.1 Request an RCS boron concentration sample analysis every interval. ______

1.1.2.2 If the boronometer is available, obtain a boronometer reading. ______

1.1.2.2.1 Ensure by indicated valve position and absence of alarms that the boronometer has sample flow. ______

1.1.2.3 Make a Unit Control Operator Log entry showing that the indicated and analyzed RCS Boron Concentration is equal to or greater than the previous RCS sample analysis. ______

1.1.2.4 If a reduction in RCS Boron Concentration is indicated, take action per S023-3-5.19, "Loss of Boron/Dilution Accident." ______

1.2 <u>When</u> the Boron Dilution Detection actions have been initiated, <u>then</u> continue to perform the subsequent actions of this instruction. ______

1.3 <u>When</u> both Startup Channels are operating, <u>then</u> notify the Shift Supervisor and terminate use of this attachment. ______

Table 1

Monitoring Frequencies for Back up Boron Dilution Detection

Maximum Volume Addition Before Monitoring Is Required	Number of Charging Pumps			
	0	1	2	3
10,000 gallons	12 hours	8 hours	4 hours	2 hours

ACTIONS ON TURBINE TRIP

INITIALS

1.0 ACTIONS

1.1 Depress the Turbine Emergency Stop Pushbutton, locking in the trip circuit, thus providing a redundant signal to actuate the various turbine trip functions. ______

1.2 Verify the following actions occur after the turbine trip:

1.2.1 "Turbine Trip Relay Tripped" (99A24) alarm is received. ______

1.2.2 High Pressure Stop Valves close. ______

1.2.3 Low Pressure Stop Valves close. ______

1.2.4 Turbine Governor Valves close. ______

1.2.5 HV-8243 A, B, C, D, E and F, Main Steam Line Drain Trap Bypasses (HS-8243) open. ______

NOTE: These valves may be closed after 10 second delay from HS-8243 to prevent excessive RCS cooldown.

1.2.6 Bled Steam Supply Valves HV-2712A and B (HS-2712) close. ______

1.2.7 The Extraction Steam Block Valves close:

1.2.7.1 HV-8800 1st Pt. Extr. to 1st Pt. Htr. (HS-8800). ______

1.2.7.2 HV-8804 1st Pt. Extr. to 1st Pt. Htr. (HS-8804). ______

1.2.7.3 HV-8808 2nd Pt. Extr. to 2nd Pt. Htr. (HS-8808). ______

1.2.7.4 HV-8806 2nd Pt. Extr. to 2nd Pt. Htr. (HS-8806). ______

1.2.7.5 HV-8812 3rd Pt. Extr. to 3rd Pt. Htr. (HS-8812). ______

1.2.7.6 HV-8810 3rd Pt. Extr. to 3rd Pt. Htr. (HS-8810). ______

1.0 ACTIONS (Continued) INITIALS

1.2.7.7 HV-8820 4th Pt. Extr. to 4th Pt. Htr. (HS-8820). ______

1.2.7.8 HV-8816 4th Pt. Extr. to 4th Pt. Htr. (HS-8816). ______

1.2.8 The reheater live steam supply valves close:

1.2.8.1 HV-2703 Reheater live steam warming isol. (HS-2703). ______

1.2.8.2 HV-2704 Reheater live steam warming isol. (HS-2704). ______

1.2.8.3 HV-2702 Live Steam to reheaters (HS-2702). ______

1.2.8.4 HV-2721 Live Steam to reheaters (HS-2721). ______

1.2.9 The turbine speed starts decreasing from 1800 rpm. ______

1.2.10 The LP Spray System functions to hold TV-2819 open for two minutes following the trip. ______

1.2.11 As the turbine coasts down verify operation of the Jacking Oil Pumps, Turning Gear Motor, and Turning Gear Clutch and the Extraction Traps per SO23-10-2, "Turbine Shutdown." ______

VERIFICATION OF RCS PRESSURE BOUNDARY

1.0 VERIFICATION

INITIALS

1.1 On PR-0100A/B, check that pressurizer pressure is trending to 2250 psia. ________

1.1.1 If pressurizer pressure is not trending to between 2225 psia and 2275 psia, then attempt to establish and maintain the required pressurizer pressure per SO23-3-5.17, "Pressurizer Pressure Control System Malfunction. ________

1.1.2 If pressurizer pressure decreases to 1825 psia, then verify actuated or actuate SIAS and CIAS and at least 5 seconds after all CEAs have been inserted stop all operating reactor coolant pumps. ________

1.2 On LR-0110A/B, check that pressurizer level is trending to 33.0% level (programmed setpoint for no load Tavg). ________

1.2.1 If pressurizer level is not trending toward 33.0% level, then attempt to establish and maintain the required level per SO23-3-5.24, "Loss of Pressurizer Level Control." ________

1.3 Check that the pressurizer safety valves indicate closed. ________

1.3.1 Use the acoustic flow indications and alarms in combination with Quench Tank indications and alarm. ________

1.4 On LI-0226A check that the Volume Control Tank (VCT) level is between 37% and 51%. ________

1.4.1 If VCT level is not between 37% and 51%, then verify the makeup system is functioning properly per SO23-3-2.2, "Makeup Operations." ________

1.5 On TI-9903-1 and TI-9911-2 check that the containment temperature is between 80°F and 120°F. ________

1.5.1 If containment temperature is not between 80°F and 120°F, then verify the containment normal heat removal system is functioning properly per SO23-1-4, "Containment Normal Heat Removal." ________

1.6 On PI-0351-1, PI-0351-2, and PI-0351-3 and PI-0351-4 check that the containment pressure is between +1.5 psig and -0.3 psig. ________

1.0 VERIFICATION (Cont'd)

INITIALS

1.6.1 If containment pressure increases to 2.8 psig, then verify actuated or actuate CIAS, SIAS, and CCAS. ________

1.6.1.1 If CIAS is actuated, then restore CCW Non-Critical Loop per S023-3-5.26.1, "Loss of Reactor Coolant Pump Component Cooling Water." ________

1.7 On LI-5853-1 and LI-5853-2 check that the containment normal sump level is not increasing. ________

1.8 At Radiation Control Panel L-103 check that the containment activity is within limits. ________

Indicator		Limits
RI-7804A1 RI-7807A2	&	Iodine - less than 4.6 X 10^2CPM
RI-7804B1 RI-7807B2	&	Particulate - less than 5.7 X 10^2CPM
RI-7804C1 RI-7807C2	&	Gaseous - less than 8.0 X 10^5CPM

1.9 If any parameter (in Step 1.1 through 1.8 above) fails to respond as indicated or does not return to its specified range after completion of prescribed action, then go to attached Figures 1 and 2 to identify the event in progress. ________

1.9.1 If use of another Emergency Instruction is indicated, in conjunction with "Emergency Plant Shutdown", then perform all applicable steps concurrently with the steps in this instruction. ________

1.9.2 If another Emergency Instruction is not indicated and SIAS was activated, then complete S023-3-5.15, "Recovery from an Inadvertant SIAS," performing all applicable steps concurrently with this instruction. ________

ACTIONS FOR EXCESSIVE COOLDOWN

INITIALS

1.1 If there is undesirable cooldown of the RCS (Tavg. below 540°F and decreasing) caused by excessive main feedwater addition, then reduce feed rate as follows:

1.1.1 On HIC-1105 and HIC-1106, use the manual control knob to match the manual output indicator with the auto output indicator. ________

1.1.2 Transfer the auto/manual transfer switch to manual. ________

NOTE:

If rapid filling of one or both steam generators has occurred, then the narrow range level indication may be indicating an erroneously low level.

1.1.3 Avoid inadvertently spilling over the can deck which may cause recirculation of relatively cold water into the tube bundle area by maintaining wide range steam generator level below 80%, until Tavg is above 530°F. ________

1.1.4 Using the manual control knob, position the valves to establish a gradual (less than 2% per minute) level recovery to ~65% steam generator narrow range level indication. ________

1.2 If there is an excessive cooldown of the RCS (Tavg below 535°F and decreasing) caused by excessive auxiliary feedwater addition, then reduce the number of operating auxiliary feedwater pumps as follows: (Listed in order of preference).

1.2.1 If P-140 is supplying both steam generators, then override and stop P-141 and P-504. ________

1.2.1.1 If steam generator level cannot be maintained, then start P-141 and/or P-504 to establish adequate flow. ________

1.2.2 If both P-141 and P-504 are supplying their respective steam generators, then override and stop P-140. ________

1.2.2.1 If steam generator level cannot be maintained, then start P-140. ________

INITIALS

1.3 Refer to Attachment 7 and verify Turbine Trip conditions are met. ________

1.4 Check the following plant equipment and verify closed or close:

1.4.1 Steam Bypass Control Valves (S023-3-2.18) HV-8423, 8424, 8425 and 8426 (HIC-8423, 8424, 8425 and 8426). ________

1.4.2 Atmospheric Dump Valves (S023-3-2.18.1) HV-8419 to S.G. E-008 and HV-8421 to S.G. E-089. ________

1.4.3 Main Feedwater Pump Before Seat Drains (S023-2-1) HV-8612/8213 to K-006, HV-8617/8226 to K-005. ________

1.5 Check the Secondary System for excessive steam leaks, etc, (i.e., Safeties Lifted, Feedwater Heater Relief Valves Lifted, etc.). ________

ACTIONS FOR EXCESSIVE COOLDOWN

1.1 If there is undesirable cooldown of the RCS (Tavg. below 540°F and decreasing) caused by excessive main feedwater addition, reduce feed rate as follows:

1.1.1 On HIC-1105 and HIC-1106, use the manual control knob to match the manual output indicator with the auto output indicator. ______

1.1.2 Transfer the auto/manual transfer switch to manual. ______

NOTE:

If rapid filling of one or both steam generators has occurred, the narrow range level indication may be indicating an erroneously low level.

1.1.3 Avoid inadvertently spilling over the can deck which may cause recirculation of relatively cold water into the tube bundle area by maintaining wide range steam generator level below 80%, until Tavg is above 530°F. ______

1.1.4 Using the manual control knob, position the valves to establish a gradual (less than 2% per minute) level recovery to ~65% steam generator narrow range level indication. ______

1.2 If there is an excessive cooldown of the RCS (Tavg below 535°F and decreasing) caused by excessive auxiliary feedwater addition, reduce the number of operating auxiliary feedwater pumps as follows: (Listed in order of preference).

1.2.1 If P-140 is supplying both steam generators, then override and stop P-141 and P-504. ______

1.2.1.1 If steam generator level cannot be maintained, restart P-141 and/or P-504 to reestablish adequate flow. ______

1.2.2 If both P-141 and P-504 are supplying their respective steam generators, then override and stop P-140. ______

1.2.2.1 If steam generator level cannot be maintained, restart P-140. ______

3.0 Longer Term Actions

3.1 Safety Injection (SI) Miniflow Valve Modifications

As discussed in SCE letters dated December 29, 1982 and January 14, 1983, the automatic RAS has been removed from the four SI miniflow isolation valves, and operating instructions have been modified to require manual closure of these valves prior to a need for the recirculation mode of operation. It is SCE's intent to leave these valves in this configuration until completion of our efforts to systematically evalute the merit of the present RAS automation with respect to overall ESF system reliability. This effort is discussed in Section 3.3.

3.2 Connector Modification Schedule

The modifications to the J3109 connectors in the PPS channels A and D as described in Section 1.1 of this report are planned to be completed during the first cold shutdown (Mode 5) of sufficient duration to accomplish the change. As stated in SCE's January 14, 1983 letter, this modification is expected to take a minimum of five days. In any case, the modification will be accomplished no later than first refueling of both units. SCE considers this proposed schedule to be consistent with the actions taken to date to minimize the chance of a momentary disconnection causing an actuation. To recapitulate these actions, SCE verified the tightness and integrity of all PPS power supply and cable connections, and in particular the Channel A and D J3109 connectors, and administrative controls as well as locking devices have been implemented to restrict access to the PPS cabinets. Access is allowed only after the senior reactor operator in the control room has been alerted, and only for monthly surveillance or for a specific maintenance procedure. Consequently, SCE believes that the Channel A and D J3109 connectors do not consititute a credible single failure in the time prior to implementing the above design change.

In addition, SCE considers that the present configuration of the connectors meets GDC-35 and that the additional administrative actions discussed above enhance this position during the interval prior to making the modification. The detailed GDC-35 discussion is provided in Section 1.3 of this report.

3.3 Future RAS Study

The objective of this task is to evaluate the merit of the present RAS automation with respect to overall ESF system reliability. The automatic RAS evaluation will investigate multiple failure scenarios which could cause concurrent false actuations and will also investigate the feasibility of corrective actions, hardware and procedures which would be appropriate to reduce the potential for such events. In addition, system modifications will be considered which have the potential to mitigate the consequences of a presumed false actuation signal. System modification such as a permissive time-delay between SIAS and RAS, Manual RAS on one train, and energize-to-actuate-RAS are typical of the approaches to be considered. It is expected that this study will be completed during the third quarter of 1983.

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ENCLOSURE II

ENCLOSURE II

RESPONSES TO ADDITIONAL NRC QUESTIONS
ENGINEERED SAFETY FEATURES ACTUATION SYSTEMS (ESFAS)
SAN ONOFRE UNITS 2&3

1 - Question

Do the surveillance procedures include specific steps which can detect "degraded" power on a Vital Bus (to PPS), e.g. reduced voltage present (or had occurred), noise spikes.

Response

Routine surveillance procedures of both Vital Bus and PPS. Power supplies involve RMS voltmeter readings only. Thus, degraded modes which change RMS output can be detected while postulated degradations of waveshapes, etc., cannot be detected.

The following alarms are available to allow the operator to detect degraded power on a Vital Bus:

Low DC Voltage - An alarm condition will be present if the Inverter DC Input Voltage falls to 110 VDC. At this time the DC Voltage normal light will go out. The alarm will be terminated when the Inverter DC Input Voltage rises above 122 volts and the alarm reset switch is pressed.

Low Air Flow - An alarm condition will be present if the Inverter Air Inlet Filters become restricted with dust or if a fan should fail. At this time the Low Air Flow Light will be lit. The alarm will terminate when the restriction is cleared or the fan is replaced and the alarm reset switch is pressed.

High Inverter Output Voltage - An alarm condition will be present if the Inverter AC Output Voltage rises to 125 volts. At this time the High Inverter Output Voltage light will be lit. The alarm will terminate when the Inverter AC Output voltage falls to 122 volts and the alarm reset switch is pressed.

Inverter Failure - An alarm condition will be present if Inverter AC Output voltage is not present. At this time the Inverter Failure Light will be lit. The alarm will be terminated when the Inverter AC Output Voltage is restored and the alarm reset switch is pressed.

Inverter Overload - An alarm condition will be present if the Inverter Output Current rises to 200 amps or higher for at least 10 msec. At this time the Inverter Overload Light will be lit. The alarm will be terminated when the Inverter Output Current falls below 190 amps and the alarm reset switch is pressed.

The relays used in the above alarm systems are Schrack RM 202610 relays and have a response time of 15 msec. However, this time does not take into account the electronics associated with each alarm.

2 - Question

Do the surveillance procedures include specific steps which can detect "degraded" power at the output of a dc power supply within the PPS?

Response

The DC power supply trouble annunciation is not designed to annunciate a degraded power condition. The power supply trouble annunciation indicates gross loss of DC power supply output to asist the operator to readily localize a power problem within the system.

When a power supply trouble annunciation has been received power has degraded to the level where circuitry in the system was actuated. However, the actuation of this circuitry will not cause a full system actuation since the degradation or loss occurs within one channel.

As indicated in the response to question 1, routine surveillance procedures of Vital Bus and PPS power supplies involve RMS voltmeter readings only and the routine monthly surveillance testing of power supplies provides information on power supply performance.

Power Supply Trouble Annunciation

All power supplies in the system provide annuncation when DC output is lost. Annunciator circuits consist of a relay on the output of each supply. The relays drop out on supply failure. Depending on the supply voltage and annunciator relay type, relays drop out between 0.225 and 0.9 volts or greater with a release time between 1.5 and 7.5 msec. The time response of the Plant Annunciator must be added to these figures. Power supply trouble annunciator relays do not seal-in; this function is provided by the Plant Annunciator System.

Matrix Power Supplies

Should vital power to one channel of the PPS be lost, or the loss or degradation of a matrix power supply occur, the condition is detectable by (a) power supply trouble annunciation (b) dropout (extinguishing) of matrix relay indicators, on the matrix test module, (c) dropout (extinguishing) of trip path indicators on the PPS local status panel, the PPS remote reactor trip status panel (in control room), the remote control modules (in the control room), and (d) trip path indicators extinguishing on the ESFAS Auxiliary Relay Cabinet (ARC) control panels. Again no single loss of a vital bus, or power supply within one channel will cause a full system actuation.

Any of the indications listed above will indicate a power supply (vital bus or DC output) problem when the degree of degradation causes circuitry to actuate in the system.

Trip Path Power Supplies

Should vital power to one channel of the PPS be lost, or the loss or degradation of a trip path power supply occur, the condition is detectable by (a) power supply trouble annunciation (b) extinguishing of a single trip path indicator on the PPS local status panel, the PPS remote reactor trip status panel (in the Control Room), the remote control modules (in the Control Room) and (c) trip path indicators extinguishing on the ESFAS ARC control panels.

Again, no single loss of a vital bus, or power supply within one channel will cause a full system actuation.

Any of the indicators listed above will indicate power supply (vital bus or DC output) problems when the degree of degradation causes cicuitry to actuate in the system (e.g., trip path relays de-energize by 3 VDC or greater).

Bistable and Bypass Power Supplies

Should vital power to one channel of the PPS be lost, or the loss or degradation of a bistable or bypass power supply occur the condition is detectable by the power supply trouble annunciation. No other indication or annunciation occurs because bistable and bypass power supplies are auctioneered across two channels. The system will operate normally, since no system circuitry, other than power supply annunciation is affected.

3 - Question

List loads on 1E buses which are shed by SIAS. Discuss consequences of spurious shedding when reactor is at full power.

Response

Section 2.2.3 of the report included as Enclosure I of this submittal provides a detailed discussion of the loads on IE buses which are shed by SIAS including the consequences of spurious shedding when the reactor is at full power.

4 - Question

Since "loss of power" (to PPS) annunciator did not occur during the 12/17/82 event, discuss the adequacy of this feature (including value of setpoint, time response, and seal-in features).

Response

The PPS power supply trouble annunciator was designed to provide indication to the operator upon deenergization of a power supply. It's purpose is to be an aid to the operator in evaluating other indications received from the PPS as a result of a power supply deenergization. The intent of this annunciator has never been to provide indication of dc surges. As discussed previously (question 2) degradation of a power supply output due to power line noise or power supply failure would be annunciated to the operator only when the degradation reaches a level causing actuation of circuitry. However, this circuitry actuation would not cause a full system actuation since the degradation occurs only within one channel, consistent single failure criteria. Also, the monthly surveillance testing is provided to assure that power supply degradation is not occuring. Therefore, the power supply trouble annunciator is adequate as designed.

5 - Question

Since power supplies (e.g. PS-32) are located on the inside surfaces of the PPS doors, what action does licensee propose to protect and surveil these components, e.g. plexiglass cover, noise monitors with fast seal-in?

Response

Protection of PPS power supplies will be achieved by administrative controls and locking devices to restrict access to the PPS cabinet as discussed in previous letters. (Access to cabinets is allowed only after the senior reactor operator in the control room has been alerted.) No changes to the surveillance procedures are planned, although a one-time program of testing of these power supplies (decay times, noise filtering, etc.) is in progress and all power supply screw connections and connectors have been inspected and tightened.

Access to the PPS power supplies is limited primarily to surveillance testing and then only when power supply adjustments and/or monitoring is required. Access is also limited to one channel at a time. The power supplies are mounted to the inside of a swing panel located behind the cabinet rear doors. To obtain access to these power supplies, the PPS cabinet door keys must be obtained from the shift supervisor. Upon opening the rear cabinet door, the power supply panel locking screws must be removed and the door swung open and locked in position. Once this has been accomplished, the technician now has access to the PPS power supplies. In order to protect the power supply terminations from damage, a plexiglass plate covers the face of each power supply. This helps to prevent inadvertent shorting of power leads.

6 - Question

Are there any design features that will allow low/noisy power coming into the PPS and outputs of dc power supplies to be detected/alarmed; prior to an ESFAS actuation?

Response

PPS channel independence by design affords detection/annunciation of low or noisy power either coming into the PPS or output from the dc power supplies prior to ESF System Actuation. Degradation will be detected when it reaches a level sufficient to affect the normal operation of the circuitry within that channel, resulting in indication and annunciation to the operators. However, power failure in a single channel (single failure criteria) will not result in full actuation of any ESF Systems.

7 - Question

Explain why during the event, the SIAS-permissive lights (red, yellow, green, blue-upper left corner) did not extinguish; why did SIAS actuate w/o this permissive condition.

Response

The SIAS Permissive Indicator operated normally during the event. The lamps are part of a non-latching circuit that indicates the state of the SIAS matrix relay contacts in the SIAS trip paths. While the trip path relays for SIAS lockout upon actuation, the matrix relay contacts within the trip path do not. This permissive signal to CSAS trip path from the SIAS trip path responds to the state of these matrix relay contacts. This permissive allows CSAS initiation only after an SIAS initiation. If the SIAS matrix relays deenergize and remain deenergized, then the SIAS permissive indication will remain extinguished. However, if the SIAS matrix relays deenergize momentarily, the SIAS trip paths will actuate the lock-out, but, the SIAS permissive will only momentarily extinguish.

8 - Question

In view of the consequences of a spurious actuation of ESFAS (downstream of instr. bistables), is it prudent to modify ESFAS to cause reactor trip directly upon ESFAS actuation? If not, why.

Response

The spurious actuation at San Onofre Unit 3 was caused by to independent failures. The PPS is designed such that no single failure will prevent or initiate a full RPS or ESFAS actuation. The design also assures that an RPS actuation occurs whenever SIAS actuates. To modify the design to cause a reactor trip on ESFAS actuation is redundant with the present design features. In addition, review of the system has indicated that a spurious actuation of SIAS without an RPS trip does not result in a safety problem.

9 - Question

The CE review of 12/17/82 event concentrated on hardware interconnections. Provide additional informaticn on functional interdependency.

Response

The CE review of the 12/17/82 event has been carried out by a multi-disciplined group of design engineers. Several specific aspects of the event have been reviewed. The initial phase of the review focused on cause and the efforts were concentrated on Plant Protection System (PPS) hardware.

As the review efforts continued, the scope increased to include the impact on the plant should a similar event occur from various initial plant conditions, including power operations. The expanded review has addressed the reactor coolant system transients and has examined the operation of systems which have equipment controlled by ESFAS signals to determine the potential for adverse effects.

The early reports and meetings with the Nuclear Regulatory staff did emphasize the hardware aspects. The January 14, 1983 letter to the NRC provided summary results of the expanded review included as Enclosure I of this submittal.

10 - Question

If one vital bus is de-energized (immediately following an inverter failure) and then a voltage spike or loss occurs on another vital bus, will full ESF actuation occur; will reactor trip occur.

Should a time limit on a vital bus being de-energized and/or off battery power be established.

Response

The scenario suggested in this question is discussed in two parts as follows:

1) Deenergization of any two vital buses will cause initiation of the RPS and various ESF Systems. Upon deenergization of the vital buses, power is lost to the process instrumentation resulting in PPS process input signals falling to zero volts. Therefore, each PPS bistable which actuates on a decreasing input signal (e.g., low pressurizer pressure, low S.G. level, etc.) will assume a tripped condition. Since this occurs in two channels the two-out-of-four coincidence will be satisfied resulting in a full actuation. However, the bistables which actuate on an increasing process input signal (e.g., high containment pressure) will not trip. Also, the PPS is designed such that deenergization of channels A and C or Channels B and D will not cause an ESF actuation. Therefore, loss of power to the specific channels noted above in conjunction with the system which has only increasing trip functions (i.e., CSAS) will not result in a full actuation. Also, because the logic associated with the initiation of EFAS is provided at the bistable level, actuation of EFAS may not occur.

2) The second part of this question is similar to the first, however, a surge is postulated on the second vital bus instead of deenergization. During the licensing of ANO-2 and San Onofre Units 2 and 3, the NRC expressed a concern regarding surges appearing on a single vital bus and its potential for causing an inadvertent actuation. In response to this concern, an analysis and various tests were performed to define the characteristics of the

surge and to show the immunity of the PPS to these surges. (Reference the responses to NRC questions 032.11, 032.18 in the San Onofre Units 2 and 3 FSAR). Since these tests indicated that the PPS is immune to these surges this question can be reduced to what happens upon deenergization of one vital bus.

The San Onofre Units 2 and 3 Technical Specifications presently incorporate requirements for vital bus energization. A copy of the applicable Technical Specifications are enclosed (Technical Specification Nos. 3.8.2.1 through 3.8.3.2).

11 - Question

In view of possibility of technician-induced transient and other technician errors, describe the training improvements planned for I/C staff.

Response

Routine training of I&C technicians in Technical Specification surveillances will be modified to include warnings and precautions to prevent technician induced transients during these procedures. Examples of these warnings and precautions are as follows:

1. Do not leave tools in cabinet.
2. Insure adequate lighting.
3. Make sure that test equipment has been checked out and is operational.
4. Be familiar with predicted test results before performing each step of the procedure.
5. If unexpected results occur, stop work and inform Control Operator and I&C Supervisor.
6. Obtain supervisor input as necessary.
7. Take precaution to minimize traffic in work area (e.g, rope off area).

12 - Question

The half-trip of RPS is attributed to improper voltage coming to the PPS from vital bus "A". Does SCE believe this condition persisted from 1405 - 1410 hours, or that another (separate) power disturbance occurred at 1410 hours. If two disturbances occurred on same day, what monitoring of bus voltage is being conducted to locate malfunction, either internal to inverter or an input to inverter.

Response

SCE believes that the 1405 and 1410 events were both caused by the same defective switch on Vital Bus A panel feeding the PPS. It is believed that this switch was generating a series of brief power interruptions in its degraded condition. Further monitoring of all vital buses to detect any other possible contributors is in progress, and includes special temporary high-speed voltage monitoring relays which will seal-in to indicate the existence of intermittent problems.

13 - Question

Has SCE obligated itself to comply with IEEE Standard 379 "Application of Single Failure Criterion."

Response

As indicated in Section 7.1.2.24 of the San Onofre Units 2 and 3 FSAR, the instrumentation and controls for the RPS and ESFAS conform to the requirements of IEEE Standard 379-1972, "IEEE Trial-Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems."

14 - Question

Prior to 12/17/82, did the design of the PPS satisfy IEEE 379-1977 Section 6.3.

Response

IEEE 379, 1977 Section 6.3 states the following:

"The potential for system actuation due to single failures shall be examined to determine whether such actuation would constitute an event with unacceptable safety consequences. For any such actuation thus identified as being unacceptable, the single failure criterion shall be met (that is, the Class 1E systems must not initiate the actuation as a result of any single detectable failure in addition to all non-detectable failures in the systems)."

The only areas of concern within the PPS cabinet which could cause a system actuation are the Channel A and D J3109 connectors. The criteria stated in Section 6.3 of IEEE-379 requires the examination of the system to determine if inadvertent actuation is an event with unacceptable safety consequences. If the consequences are unacceptable, then the failure which caused the event must meet the single failure criteria. However if the consequences are acceptable, nothing more is required. Since an analysis has been performed (as discussed previously in Section 2), which shows that an inadvertent actuation of the ESF does not produce unacceptable safety consequences, SCE and CE conclude that the system meets this criteria.

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3/4.8.2 D.C. SOURCES

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum the following D.C. electrical sources shall be OPERABLE:

a. 125-volt battery bank A (2B007), and its associated full capacity charger.

b. 125-volt battery bank B (2B008), and its associated full capacity charger.

c. 125-volt battery bank C (2B009) and its associated full capacity charger.

d. 125-volt battery bank D (2B010) and its associated full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

a. With one of the required battery banks inoperable, restore the inoperable battery bank to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one of the required full capacity chargers inoperable, demonstrate the OPERABILITY of its associated battery bank by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the battery inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each 125-volt battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

1. The parameters in Table 4.8-2 meet the Category A limits, and

2. The total battery terminal voltage is greater than or equal to 129-volts on float charge.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

1. The parameters in Table 4.8-2 meet the Category B limits,

2. There is no visible corrosion at either terminals or connectors, or the connection resistance of these items is less than 150×10^{-6} ohms, and

3. The average electrolyte temperature of ten connected cells is above 60°F.

c. At least once per 18 months by verifying that:

1. The cells, cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration,

2. The cell-to-cell and terminal connections are clean, tight, and coated with anti-corrosion material,

3. The resistance of each cell-to-cell and terminal connection is less than or equal to 150×10^{-6} ohms, and

4. The battery charger will supply at least 300 amperes at 125-volts for at least 12 hours.

d. At least once per 18 months, during shutdown, by verifying that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual or simulated emergency loads for the design duty cycle when the battery is subjected to a battery service test.

e. At least once per 60 months, during shutdown, by verifying that the battery capacity is at least 80% of the manufacturer's rating when subjected to a performance discharge test. Once per 60 month interval, this performance discharge test may be performed in lieu of the battery service test required by Surveillance Requirement 4.8.2.1d.

f. Annual performance discharge tests of battery capacity shall be given to any battery that shows signs of degradation or has reached 85% of the service life expected for the application. Degradation is indicated when the battery capacity drops more than 10% of rated capacity from its average on previous performance tests, or is below 90% of the manufacturer's rating.

TABLE 4.8-2

BATTERY SURVEILLANCE REQUIREMENTS

	CATEGORY A(1)	CATEGORY B(2)	
Parameter	Limits for each designated pilot cell	Limits for each connected cell	Allowable(3) value for each connected cell
Electrolyte Level	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	>Minimum level indication mark, and ≤ ¼" above maximum level indication mark	Above top of plates, and not overflowing
Float Voltage	≥ 2.13 volts	≥ 2.13 volts(c)	> 2.07 volts
Specific Gravity(a)	≥ 1.200(b)	≥ 1.195	Not more than .020 below the average of all connected cells
		Average of all connected cells > 1.205	Average of all connected cells ≥ 1.195(b)

(a) Corrected for electrolyte temperature and level.
(b) Or battery charging current is less than 2 amps when on charge.
(c) Corrected for average electrolyte temperature in accordance with IEEE Std 450-1980.
(1) For any Category A parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that within 24 hours all the Category B measurements are taken and found to be within their allowable values, and provided all Category A and B parameter(s) are restored to within limits within the next 6 days.
(2) For any Category B parameter(s) outside the limit(s) shown, the battery may be considered OPERABLE provided that the Category B parameters are within their allowable values and provided the Category B parameter(s) are restored to within limits within 7 days.
(3) Any Category B parameter not within its allowable value indicates an inoperable battery.

ELECTRICAL POWER SYSTEMS

D.C. SOURCES

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, two 125-volt battery banks and their associated full capacity chargers shall be OPERABLE.

APPLICABILITY: MODES 5 and 6.

ACTION:

a. With the required battery banks inoperable, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes or movement of irradiated fuel; initiate corrective action to restore the required battery banks to OPERABLE status as soon as possible.

b. With the required full capacity chargers inoperable, demonstrate the OPERABILITY of their associated battery banks by performing Surveillance Requirement 4.8.2.1.a.1 within one hour, and at least once per 8 hours thereafter. If any Category A limit in Table 4.8-2 is not met, declare the batteries inoperable.

SURVEILLANCE REQUIREMENTS

4.8.2.2 The above required 125-volt battery banks and chargers shall be demonstrated OPERABLE per Surveillance Requirement 4.8.2.1.

3/4.8.3 ONSITE POWER DISTRIBUTION SYSTEMS

OPERATING

LIMITING CONDITION FOR OPERATION

3.8.3.1 The following electrical busses shall be energized in the specified manner:

a. Division #1 A.C. Emergency Busses consisting of:
   1. 4160 volt Emergency Bus # 2A04
   2. 480 volt Emergency Bus # 2B04

b. Division #2 A.C. Emergency Busses consisting of:
   1. 4160 volt Emergency Bus # 2A06
   2. 480 volt Emergency Bus # 2B06

c. 120 volt A.C. Vital Bus # 2Y01 energized from its associated inverter connected to D.C. Bus # 2D1*.

d. 120 volt A.C. Vital Bus # 2Y02 energized from its associated inverter connected to D.C. Bus # 2D2*.

e. 120 volt A.C. Vital Bus # 2Y03 energized from its associated inverter connected to D.C. Bus # 2D3*.

f. 120 volt A.C. Vital Bus # 2Y04 energized from its associated inverter connected to D.C. Bus # 2D4*.

g. 125 volt D.C. Bus # 2D1 energized from Battery Bank 2B007.

h. 125 volt D.C. Bus # 2D2 energized from Battery Bank 2B008.

i. 125 volt D.C. Bus # 2D3 energized from Battery Bank 2B009.

j. 125 volt D.C. Bus # 2D4 energized from Battery Bank 2B010.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

a. With one of the required divisions of A.C. Emergency busses not fully energized, re-energize the division within 8 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

b. With one A.C. Vital Bus either not energized from its associated inverter, or with the inverter not connected to its associated D.C. Bus: (1) re-energize the A.C. Vital Bus within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours; and (2) re-energize the A.C. Vital Bus from its associated inverter connected to its associated D.C. Bus within 24 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

c. With one D.C. Bus not energized from its associated Battery Bank, re-energize the D.C. Bus from its associated Battery Bank within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

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*One inverter may be disconnected from its D.C. Bus for up to 24 hours, as necessary, for the purpose of performing an equalizing charge on its associated battery bank provided (1) its vital bus is energized, and (2) the vital busses associated with the other battery banks are energized from their associated inverters and connected to their associated D.C. Busses.

SURVEILLANCE REQUIREMENTS

4.8.3.1 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.

ELECTRICAL POWER SYSTEMS

ONSITE POWER DISTRIBUTION SYSTEMS

SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.3.2 As a minimum, the following electrical busses shall be energized in the specified manner:

a. One division of A.C. Emergency Buses consisting of one 4160-volt and one 480-volt A.C. Emergency Bus.

b. 2 - 120 volt A.C. Vital Busses energized from their associated inverters connected to their respective D.C. Busses.

c. 2 - 125 volt D.C. Busses energized from their associated battery banks.

APPLICABILITY: MODES 5 and 6

ACTION:

With any of the above required electrical busses not energized in the required manner, immediately suspend all operations involving CORE ALTERATIONS, positive reactivity changes, or movement of irradiated fuel, initiate corrective action to energize the required electrical busses in the specified manner as soon as possible.

SURVEILLANCE REQUIREMENTS

4.8.3.2 The specified busses shall be determined energized in the required manner at least once per 7 days by verifying correct breaker alignment and indicated voltage on the busses.