U. S. NUCLEAR REGULATORY COMMISSION
REGION V

Report No. 50-312/86-09

License No. DPR-54

Licensee: Sacramento Municipal Utility District
P. O. Box 15830
Sacramento, California 95813

Facility Name: Rancho Seco Unit 1

Inspection at: Meeting at Region V USNRC

Inspection Conducted: February 10, 1986

Inspectors: W. G. Albert — 2-28-86 Date Signed

Approved by: L. Miller — 2-28-86. Date Signed

Summary:

Meeting in Region V offices on February 10, 1986 (Report 50-312/86-09)

Areas Inspected: Management meeting to discuss proposed actions by licensee to be resolved prior to restart of Rancho Seco from the December 26, 1985 outage which resulted in an excessive cooldown transient. The meeting involved approximately 80 hours by various Region V personnel.

Results: This meeting did not involve direct inspection of licensee activities.

Details

1. Meeting Attendance

NRC Region V

J. Martin
B. Faulkenberry
A. Chaffee
D. Kirsch
J. Crews
W. Albert
G. Cook
A. Johnson
A. Toth
L. Miller
C. Myers
R. Pate
R. Scarano
F. Wenslawski
G. Yuhas
G. Perez

Sacramento Municipal Utility District (SMUD)

R. Rodriguez
R. Dieterich
S. Crunk
S. Redeker
J. Field
L. Schwieger
N. Brock
G. Coward
D. Whitney
J. Williams
T. Turner

NRC-NRR

J. Stolz
S. Miner

In addition to the above, the meeting was covered by channel 10 - KXTV of Sacramento, a CBS affiliate.

2. Meeting Content

The licensee made presentations as shown on the agenda included with the licensee's handout; attached to this report.

The NRC staff in turn presented the licensee with their current perception of concerns and issues which need to be resolved prior to plant restart, and emphasized that additional concerns may result from the findings of the NRC's Incident Investigation Team (IIT), the report of which is currently scheduled for presentation before the Commission on February 25, 1986.

The NRC expressed concern about the licensee's current schedule of March 8, 1986 for plant restart and stated that the schedule appeared to be very "challenging", particularly since there may be additional findings and/or recommendations resulting from the IIT's investigation. This view was expressed to licensee management at the conclusion of the meeting.

Attachments 2 and 3 are copies of NRC handouts.

Attachment 1

AGENDA

SMUD/NRC MEETING 2/10/86

RANCHO SECO TRANSIENT OF 12/26/85

1. INTRODUCTION	RON RODRIGUEZ
2. ROOT CAUSE ANALYSIS	STEVE CRUNK
3. HARDWARE "FIXES"	DAN WHITNEY
4. PLANT MODIFICATIONS	DAN WHITNEY
5. PROCEDURE CHANGES AND TRAINING	STEVE REDEKER
6. 100 DEGREE COOLDOWN LIMIT	BOB DIETERICH
7. PRESSURIZED THERMAL SHOCK	BOB DIETERICH
8. START-UP SCHEDULE	RON RODRIGUEZ

ROOT CAUSE ANALYSIS

STEVE CRUNK

DIRECT CAUSE:

THE IMMEDIATE TECHNICAL OR ENGINEERING EXPLANATION FOR THE FAILURE OBSERVED.

ROOT CAUSE:

THE PROGRAMMATIC SHORTFALL OR LACK THAT ALLOWED THE FAILURE TO OCCUR.

ROOT CAUSE ANALYSIS

COMPLEX EVENT REQUIRED BREAKDOWN
INTO FIVE "EVENT THEMES"

1. LOSS OF ICS
2. RAPID COOLDOWN
3. MAKEUP PUMP
4. HEALTH PHYSICS
5. EMERGENCY PLAN

ISSUES REVIEWED BY IAG

1. PROCEDURAL ADEQUACY
2. DESIGN FEATURES
3. HUMAN FACTORS
4. OTHER RELATED ISSUES INCLUDING ITEMS SUCH AS:
   - TRAINING
   - PREVENTIVE MAINTENANCE
   - PERSONNEL ACCESS
   - ADDITIONAL TECHNICAL ANALYSIS BY B & W

ROOT CAUSES AND CONTRIBUTORY CAUSES

(1) LOSS OF ICS	MANUFACTURING ERROR	° DESIGN CHARACTERISTICS OF MODULE ° S1, S2 SWITCH DELAY
(2) RAPID COOLDOWN	DELAY IN IMPLEMENTING DESIGN CHANGES TO MITIGATE EFFECTS OF LOSS OF ICS	° PROCEDURES ° TRAINING
(3) MAKEUP PUMPS	PROCEDURES	TRAINING
(4) HEALTH PHYSICS	HUMAN PERFORMANCE	° TRAINING ° IMPRECISE DEFINITION OF HP RESPONSIBILITIES AUTHORITIES/DUTIES
(5) EMERGENCY PLAN	TRAINING	° HUMAN PERFORMANCE ° PROCEDURES

AREAS FOR IMPROVEMENT

1. PROCEDURAL ADEQUACY
   - GUIDANCE FOR RECOVERY FROM AFAS
   - ICS CASUALTY PROCEDURE
   - GUIDANCE E.05 COOLDOWN PROCEDURE
   - CLARIFICATION AND/OR REORGANIZATION OF EMERGENCY PLAN

2. DESIGN FEATURES
   - AUX FW VALVES, TBVs, ADVs POSITION ON LOSS OF ICS

3. HUMAN FACTORS
   - UNDERSTANDING OF HP ASSIGNMENT/INVOLVEMENT
   - VISIBILITY OF ICS POWER SWITCHES

4. HUMAN PERFORMANCE
   - EMERGENCY PLAN IMPLEMENTATION
   - LIMITING COOLDOWN BY SECURING AFW PUMPS
   - ADHERENCE TO RAD-CON PROCEDURES

5. OTHER ISSUES
   - TRAINING
   - PREVENTIVE MAINTENANCE

HARDWARE "FIXES"

DAN WHITNEY

EQUIPMENT REQUIRING REPAIR FOLLOWING EVENT

1. AFW FLOW CONTROL VALVE FV-20527
2. AFW ISOLATION VALVE FWS-063
3. MU PUMP P-236 SEAL/SHAFT DAMAGE
4. ICS PSM
5. ICS WIRING CHANGEOUT
6. ICS S1/S2 TIME DELAY
7. RAD MONITOR R-15001 SAMPLE PUMP SEALS
8. WG COMPRESSOR
9. PRI/SEC LEAK

PLANT MODIFICATIONS

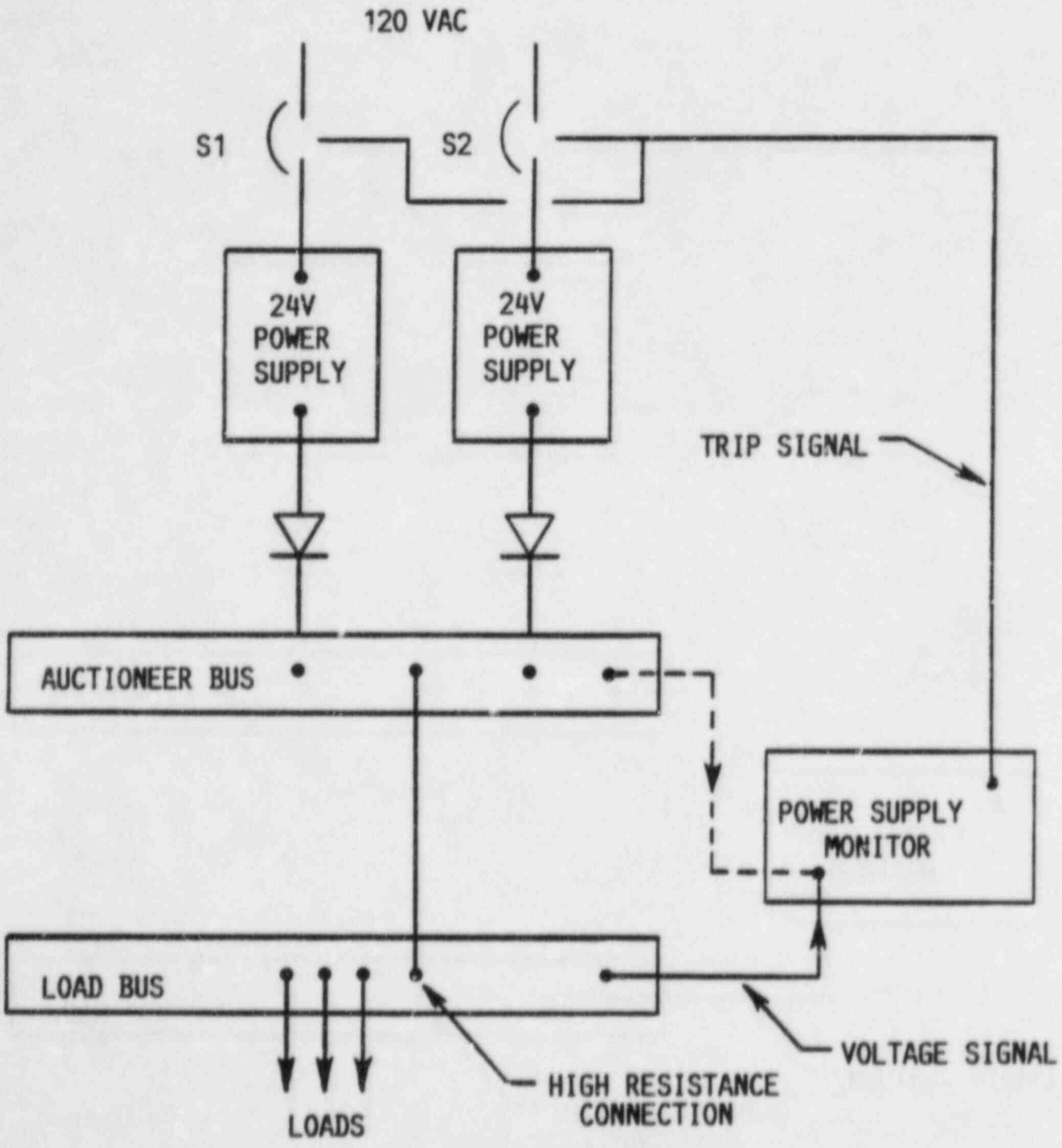
DAN WHITNEY

PLANT MODIFICATIONS

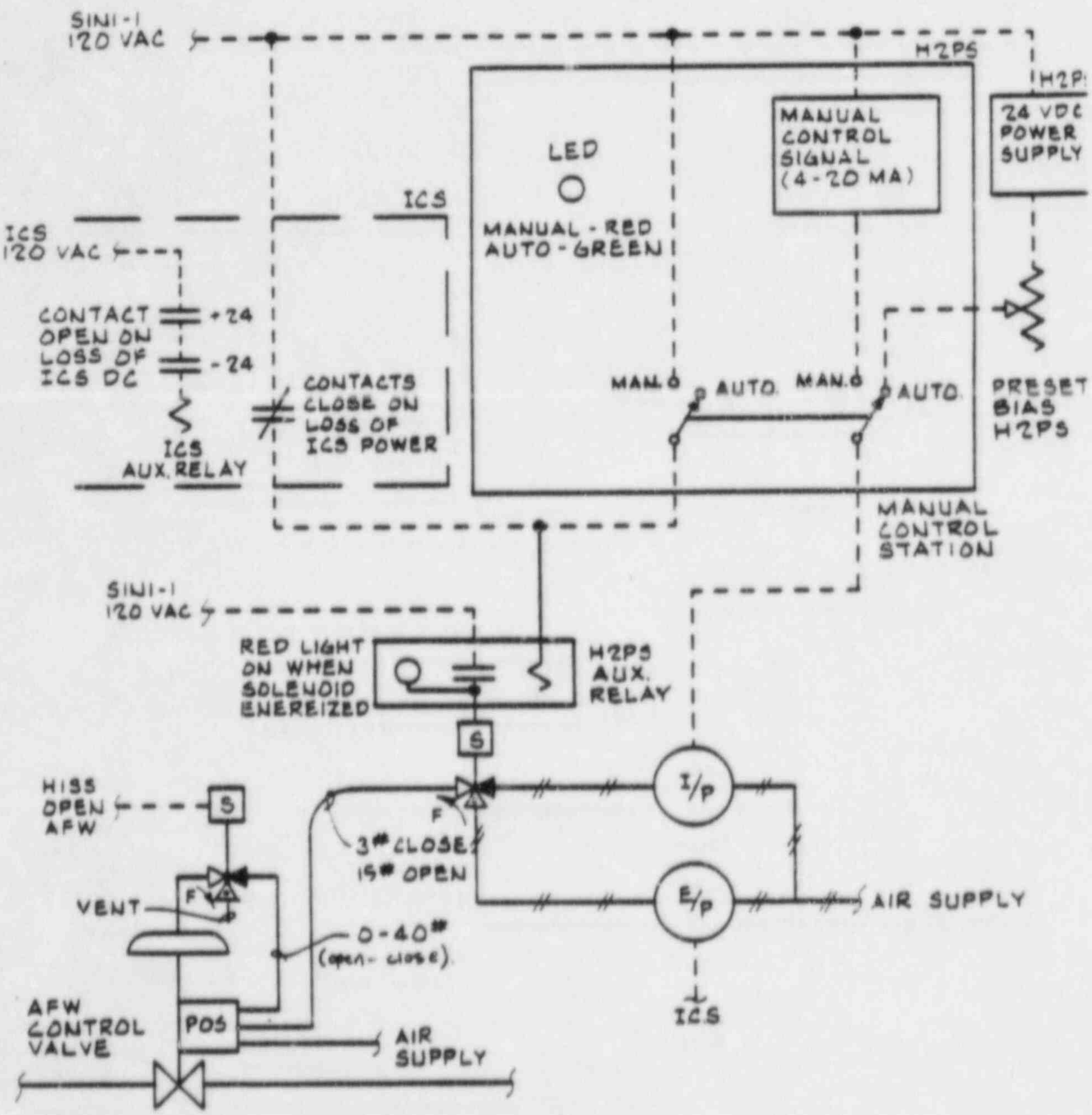
- TO PRECLUDE LOSS OF ICS POWER
- TO MITIGATE TRANSIENTS
- UNDER CONSIDERATION

MODIFICATIONS

- ICS CABINETS AND POWER SUPPLIES
- ICS/AFW FLOW CONTROL VALVE INTERFACE
- ICS/ADV INTERFACE
- ICS/TBV INTERFACE
- TECHNICAL SUPPORT CENTER (TSC) SPRINKLER SYSTEM
- MAIN STEAM LINE FAILURE LOGIC SETPOINT CHANGE

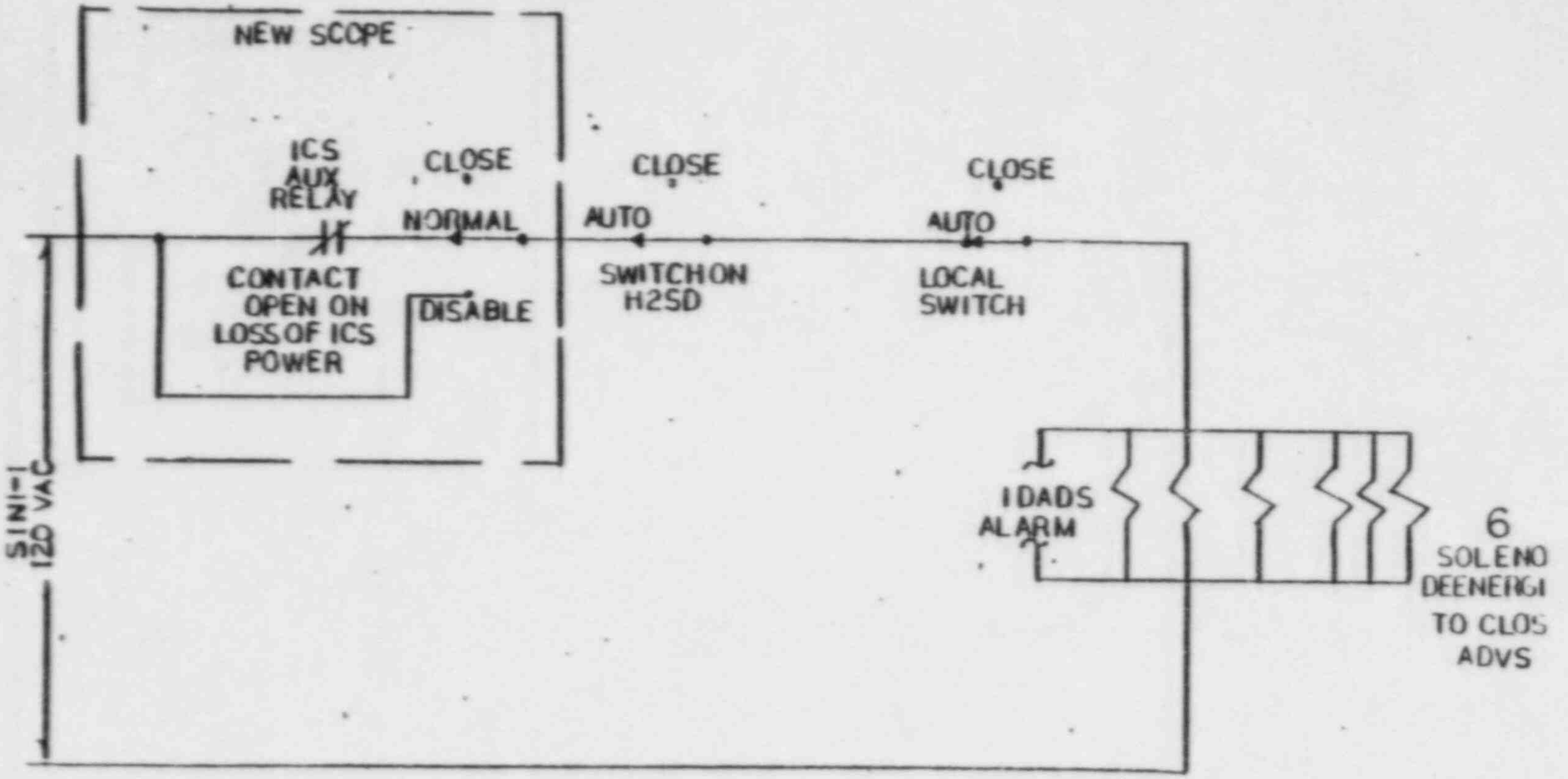


ICS POWER DISTRIBUTION



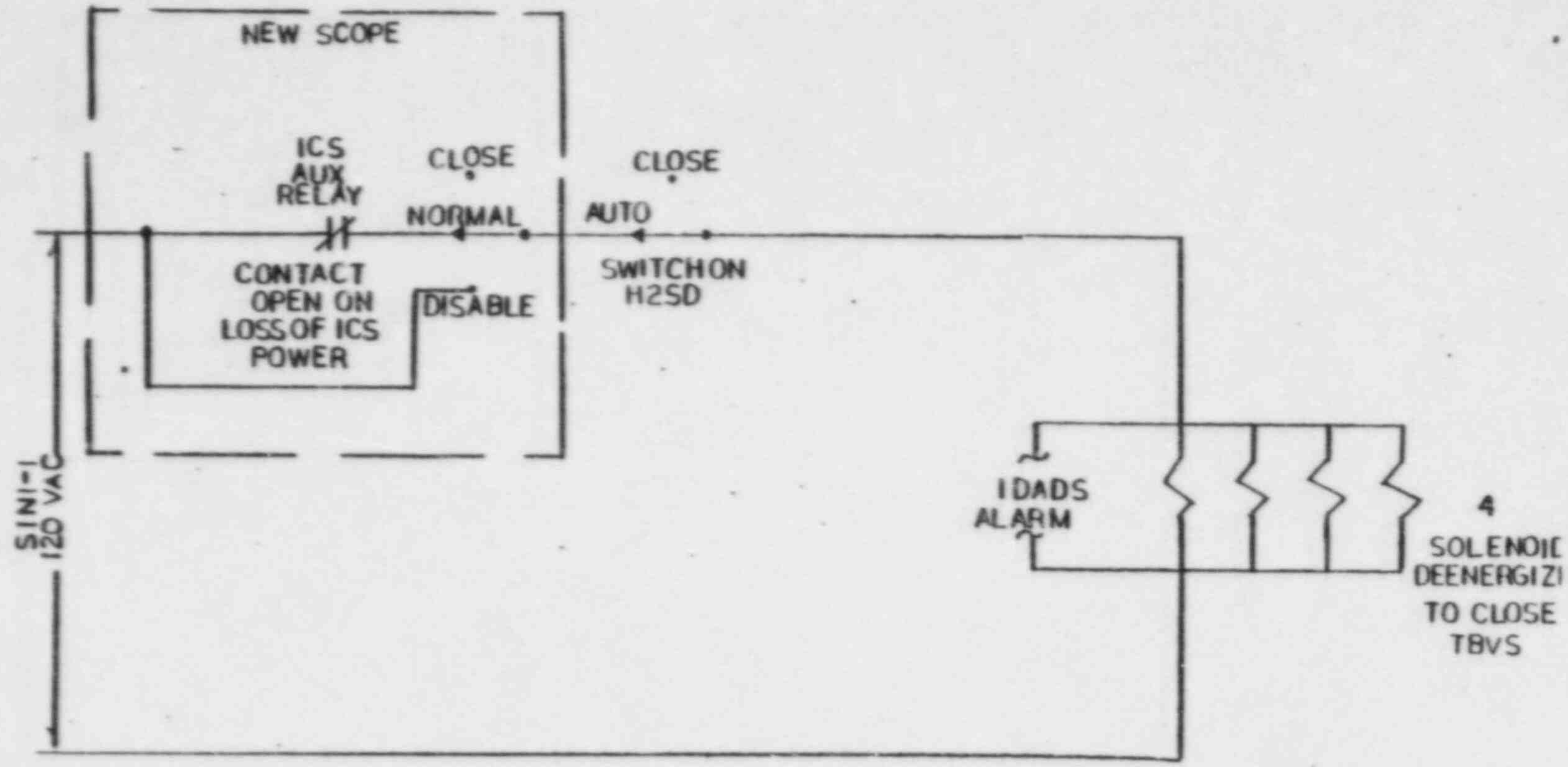
AUXILIARY FEEDWATER CONTROL
INDEPENDENT OF I.C.S.

FIGURE 1



ADV MANUAL CONTROL

FIGURE 2

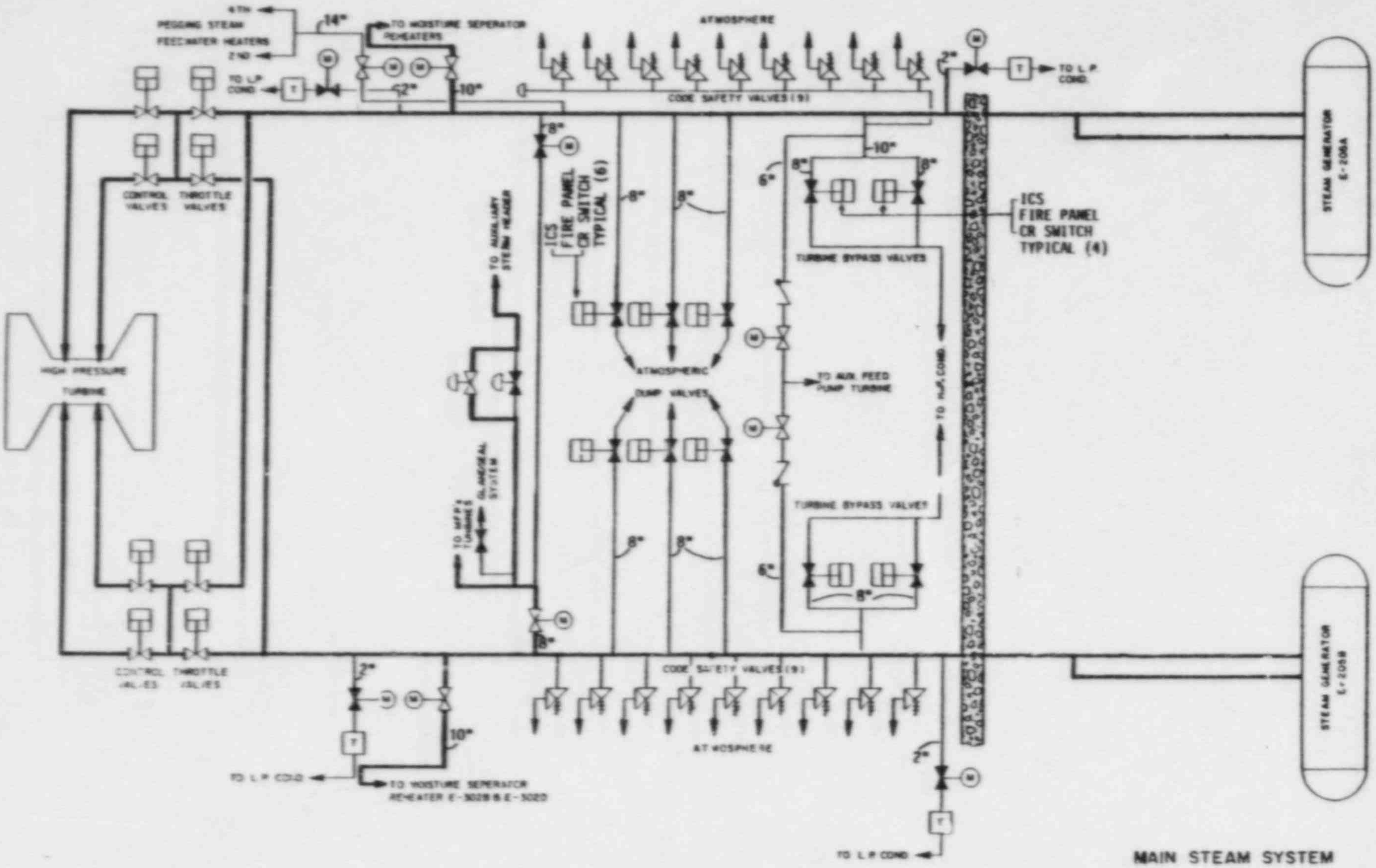


TBV MANUAL CONTROL

FIGURE 3

MODIFICATIONS CONSIDERED

- MFP TURBINE CONTROL ON LOICS POWER
- MAIN STEAM LINE ISOLATION WITHOUT ICS POWER
- INTERLOCKS ON MAKEUP/HPI PUMPS
- CR INSTRUMENT FAILURES ON LOICS
- CR NOISE FROM ESSENTIAL HVAC
- RADIATION MONITOR BLOWER ISOLATION ON SFAS



MAIN STEAM SYSTEM

SUMMARY

- IMPLEMENTING MODIFICATIONS TO REDUCE LIKELYHOOD OF LOICS POWER
- IMPLEMENTING MODIFICATIONS TO ENHANCE THE EFFECTIVENESS OF THE OPERATOR FROM THE CR
- CR CONTROL INDEPENDENT OF ICS FOR:
  - AFW FLOW VALVES
  - ADV's
  - TBV's
- SIMILAR TO OTHER B & W PLANTS

POSITIONS OF KEY VALVES AND SYSTEMS FOLLOWING LOSS OF ICS DC POWER

	RANCHO SECO	DAVIS BESSE	CRYSTAL RIVER	ANO-1	OCONEE	TMI-1
EQUIPMENT TYPE	BAILEY 820	BAILEY 820	BAILEY 820	BAILEY 820	BAILEY 721	BAILEY 721
TBPV	50%	0%	0%	50%	N/A	N/A
ADV	50%	0%	N/A	N/A	N/A	N/A
EFW CONTROL VALVES	50%	N/A	N/A	N/A	N/A	N/A
STARTUP FW FLOW PATH	50%	50%	50%	50%	N/A	N/A
MAIN FW FLOW PATH	0%	0%	0%	0%	N/A	N/A
MF PUMP GOVERNOR	2600 RPM	4690 RPM	4260 RPM	MIN SPEED	N/A	N/A
ROD CONTROL	TO MANUAL	TO MANUAL	TO MANUAL	TO MANUAL	N/A	N/A
MAIN TURBINE GOVERNOR	AS IS	AS IS	AS IS	AS IS	N/A	N/A
DOES PLANT HAVE MSIVs?	NO	YES	YES	YES	NO	YES

PROCEDURE CHANGES AND TRAINING

STEVE REDEKER

PROCEDURES

LOSS OF ICS RAPID COOLDOWN

- ABNORMAL TRANSIENT OPERATING GUIDELINE (ATOG) METHOD IS SOUND FOR LOSS OF ICS AND OVERCOOLING
- RANCHO SECO PROCEDURES CONFORMED TO ATOG TO TERMINATE OVERCOOLING
- CRITERIA WHEN TO TAKE CERTAIN ACTIONS TO TERMINATE OVERCOOLING HAVE BEEN DEVELOPED AND IMPLEMENTED
- CRITERIA FOR OTHER SYMPTOMS WERE IN PLACE
- ICS RECOVERY PROCEDURE HAS BEEN IMPLEMENTED
- MODIFICATIONS HAVE BEEN INCLUDED IN PROCEDURES

MAKEUP PUMP

- EMERGENCY PROCEDURES REVISED TO ASSURE SUCTION IS AVAILABLE
- RECOVERY FROM SFAS ACTUATION PROCEDURE IMPLEMENTED

WATCHSTANDING

- IMPROVED DIVISION OF RESPONSIBILITIES BETWEEN EMERGENCY COORDINATOR AND SENIOR CONTROL ROOM OPERATOR

EMERGENCY PLAN

- IMPROVEMENT TO ASSURE USE OF PROPER FORMS
- CORRECTIONS
- LONG TERM EFFORT TO SIMPLIFY

HEALTH PHYSICS

- IMPROVED PROCEDURE FOR ENTRY INTO AREAS OF UNKNOWN RADIOACTIVITY

PLANT TRIP RESPONSE TEAM

- DOSE ASSESSMENT
- FOLLOWUP NOTIFICATIONS
- RESTART EVALUATION
- REPAIR ACTIVITIES

TRAINING

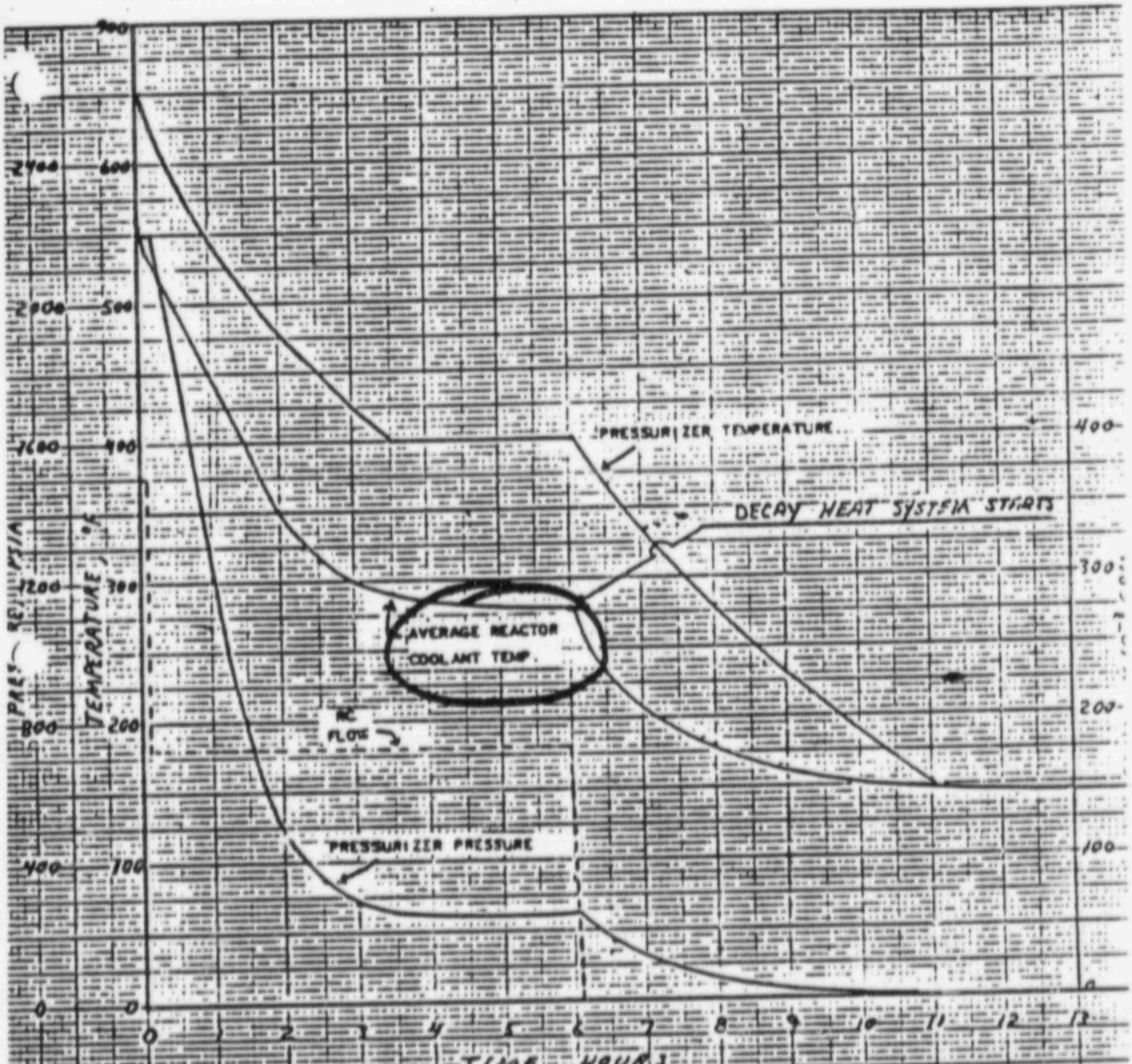
- EVENT AWARENESS
- PLANT MODIFICATIONS
- EMERGENCY PROCEDURES
- ICS RECOVERY PROCEDURE
- GENERAL PROCEDURES
- VALVE OPERATION
- WATCHSTANDING
- HEALTH PHYSICS
- EMERGENCY PLAN

<u>100 DEGREE COOLDOWN LIMIT</u>

BOB DIETERICH

NORMAL COOLDOWN DEFINED

AS 100 F/HR DESIGN



CUSTOMER SMUD	JOB NO 620-0017
SUBJECT FIGURE 18-1	
TRANSIENT NO 18 (COOLDOWN FROM 8% FP) RC AND PRESSURIZER	BY VJG
	DATE 8-28-70

Table 2

Operating Transient Cycles

Transient Number	Transient Description	Design Cycles
1	Heatup and Cooldown	
	@ 100°F/hr heatup and cooldown	120
	@ 35°F/hr heatup and 100°F/hr cooldown	120
	Total	240
2	Power change 0 to 15% and 15 to 0%	1440
3	Power Loading 8% to 100% power	48,000
4	Power Unloading 100% to 8% power	48,000
5	10% Step Load Increase	8,000
6	100% Step Load Decrease	8,000
7	Step Load Reduction (100% to 8% Power)	
	Resulting from turbine trip	160
	Resulting from electrical load rejection	150
	Total	310
8	Reactor Trip	
	Type A	40
	Type B	160
	Type C	88
	Trips included in transient numbers 11, 15 16, 17 & 21	112
	Total	400
9	Rapid Depressurization	80
10	Change of Flow	20
11	Rod Withdrawal Accident	40
12	Hydrotests	20
13	Steady-State Power Variations	-
14	Control Rod Drop	40
15	Loss of Station Power	40
16	Steam Line Failure	1
17A	Loss of Feedwater to One Steam Generator	20
17B	Stuck Open Turbine Bypass Valve	10

APPENDIX G TO 10 CFR 50 REQUIRES CALCULATION OF P/T LIMIT

- HEATUP
- COOLDOWN
- VARIOUS VALUES OF DTT
- SPECIMEN PROGRAM
- LARGE FACTORS OF SAFETY

3.1.2 PRESSURIZATION, HEATUP, AND COOLDOWN LIMITATIONS

Specification

3.1.2.1 Inservice Leak and Hydrostatic Tests:

55▶◀ Pressure temperature limits for the first eight EFP years of inservice leak and hydrostatic tests are given in Figure 3.1.2-3. Heatup and cooldown rates shall be restricted according to the rates specified in Figure 3.1.2-3.

3.1.2.2 Heatup Cooldown:

55▶◀ For the first eight EFP years of power operations, the reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-1 and Figure 3.1.2-2 respectively. Heatup and cooldown rates shall not exceed the rates stated on the associated figure.

3.1.2.3 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 130°F.

3.1.2.4 The pressurizer heatup and cooldown rates shall not exceed 100°F in any 1-hour period.

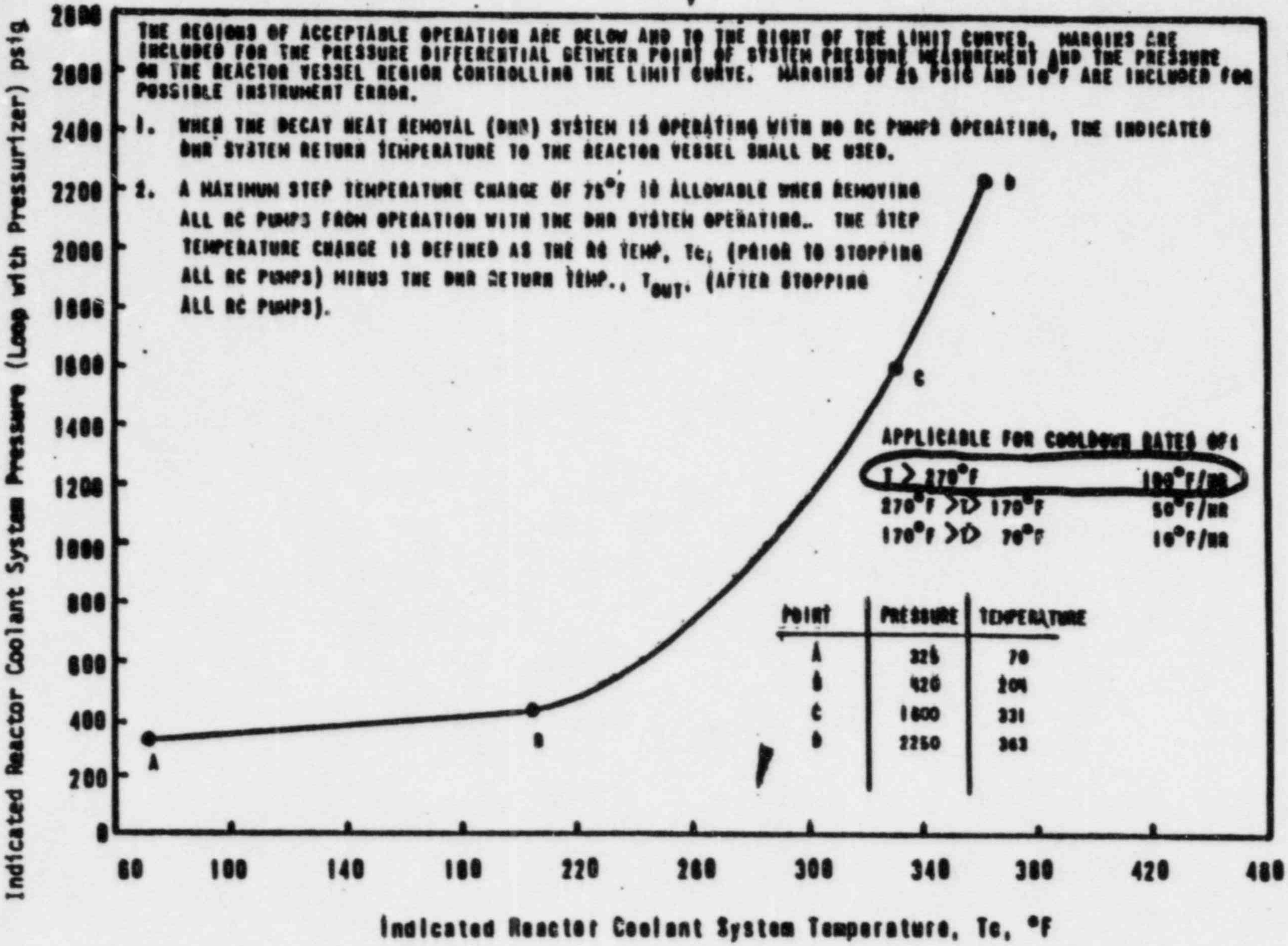
3.1.2.5 The spray shall not be used if the temperature difference between the pressurizer and spray fluid is greater than 410°F.

55▶◀ 3.1.2.6 Prior to exceeding eight effective full power years of operation, Figures 3.1.2-1, -2, and -3 shall be updated for the next service period in accordance with 10 CFR 50, Appendix G, Section V.B. The highest predicted adjusted reference temperature of all the beltline materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.7.

3.1.2.7 The updated proposed technical specifications referred to in 3.1.2.6 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR 50, Appendix G, Section V.C.

55▶◀

Figure 3.1.2-2



THE REGIONS OF ACCEPTABLE OPERATION ARE BELOW AND TO THE RIGHT OF THE LIMIT CURVES. MARGINS ARE INCLUDED FOR THE PRESSURE DIFFERENTIAL BETWEEN POINT OF SYSTEM PRESSURE MEASUREMENT AND THE PRESSURE ON THE REACTOR VESSEL REGION CONTROLLING THE LIMIT CURVE. MARGINS OF 25 PSIG AND 10°F ARE INCLUDED FOR POSSIBLE INSTRUMENT ERROR.

1. WHEN THE DECAY HEAT REMOVAL (DHR) SYSTEM IS OPERATING WITH NO RC PUMPS OPERATING, THE INDICATED DHR SYSTEM RETURN TEMPERATURE TO THE REACTOR VESSEL SHALL BE USED.

2. A MAXIMUM STEP TEMPERATURE CHANGE OF 75°F IS ALLOWABLE WHEN REMOVING ALL RC PUMPS FROM OPERATION WITH THE DHR SYSTEM OPERATING. THE STEP TEMPERATURE CHANGE IS DEFINED AS THE RC TEMP, T_{C_i} (PRIOR TO STOPPING ALL RC PUMPS) MINUS THE DHR RETURN TEMP., T_{OUT}, (AFTER STOPPING ALL RC PUMPS).

APPLICABLE FOR COOLDOWN RATES OF:

T > 270°F	100°F/HR
270°F > T > 170°F	50°F/HR
170°F > T > 70°F	10°F/HR

POINT	PRESSURE	TEMPERATURE
A	325	70
B	420	204
C	1800	331
D	2250	363

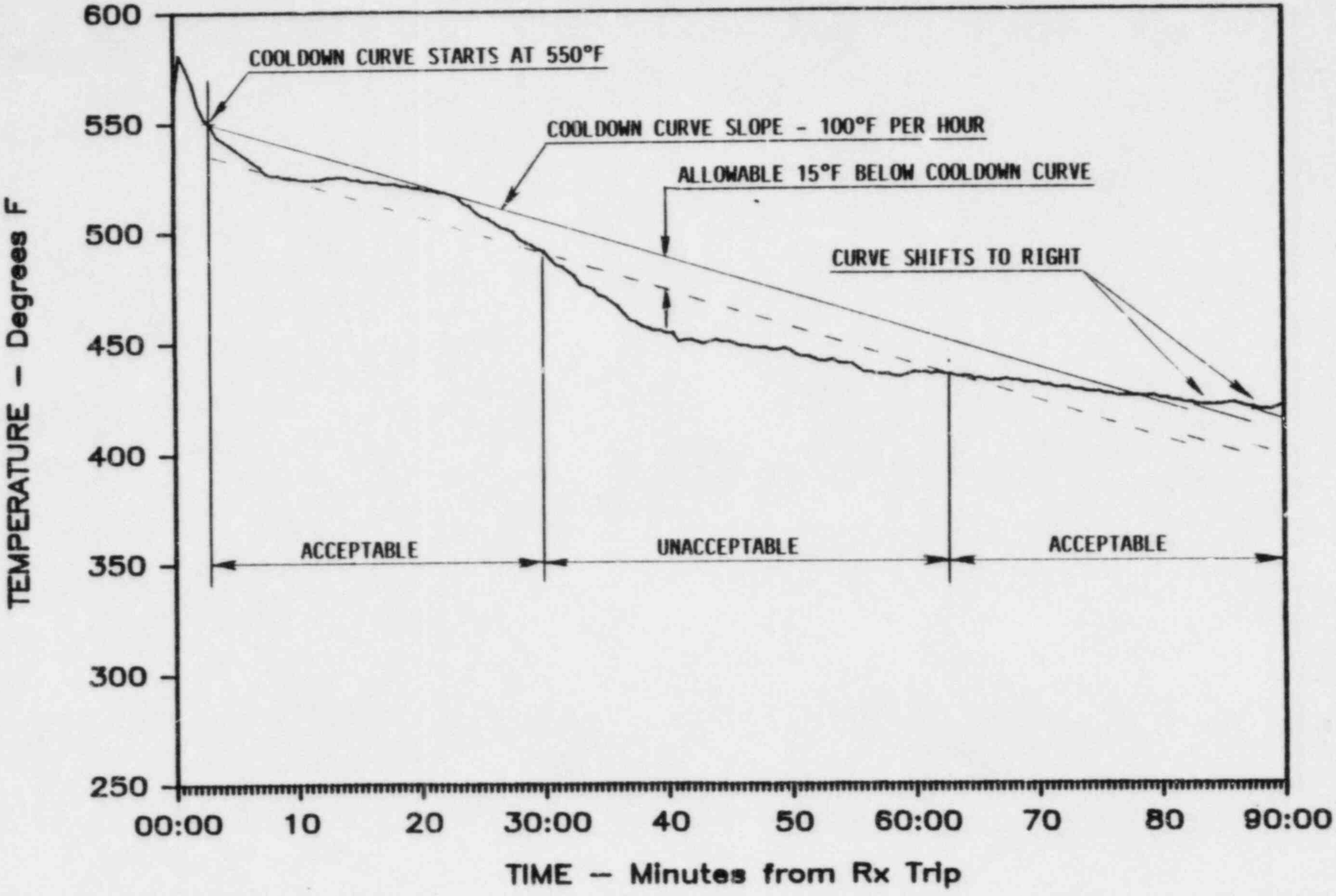
REACTOR COOLANT SYSTEM, NORMAL OPERATION-COOLDOWN LIMITATIONS, APPLICABLE FOR FIRST 8.0 EFFECTIVE FULL POWER YEARS

LINEAR COOLDOWN USED IN ANALYSIS

- THERMAL GRADIENT IS REAL LIMIT
- MANY WAYS TO COOL DOWN
- PRESENT INTERPRETATION

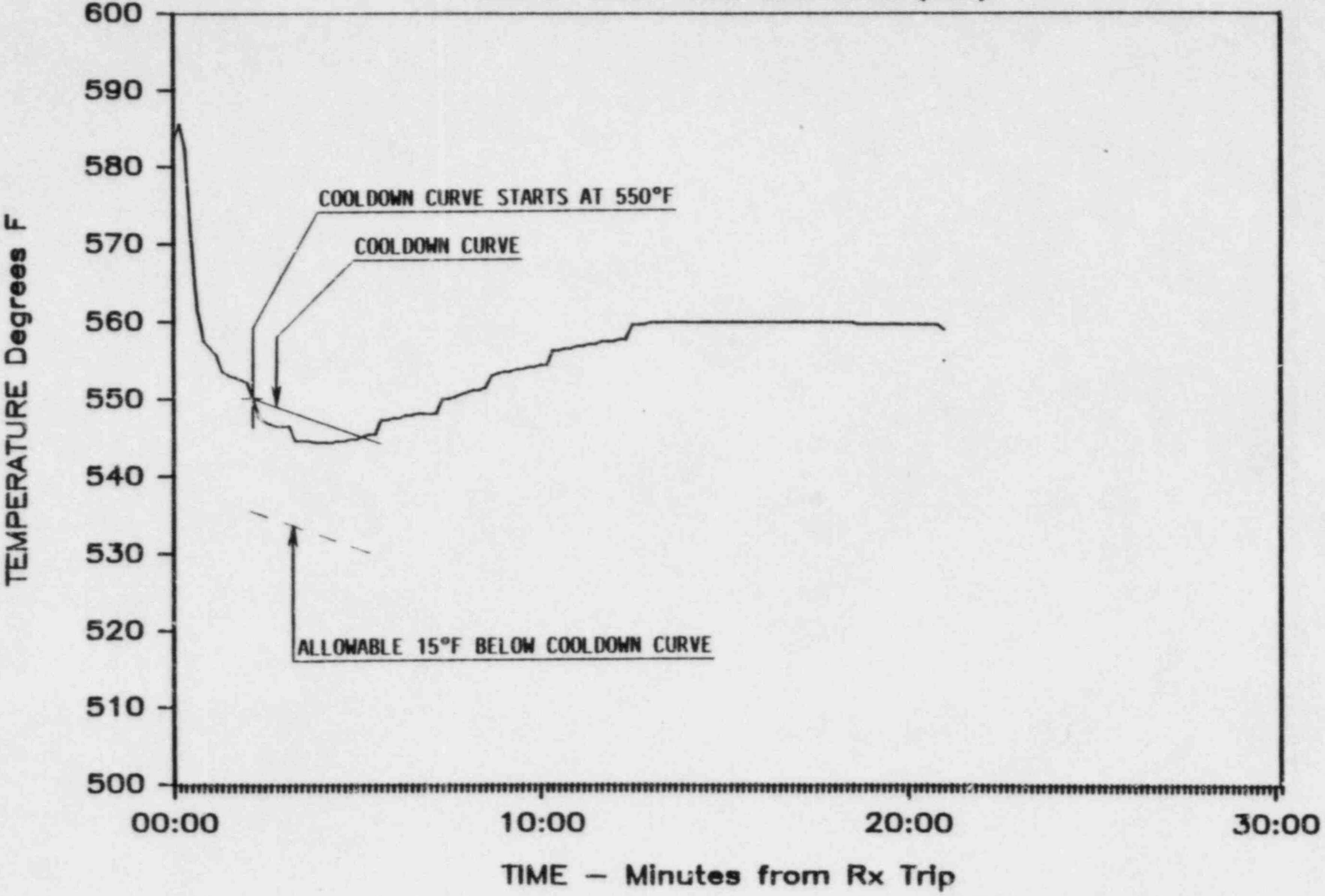
FICTICIOUS Rx TRIP

FOR EXPLANATION PURPOSES ONLY



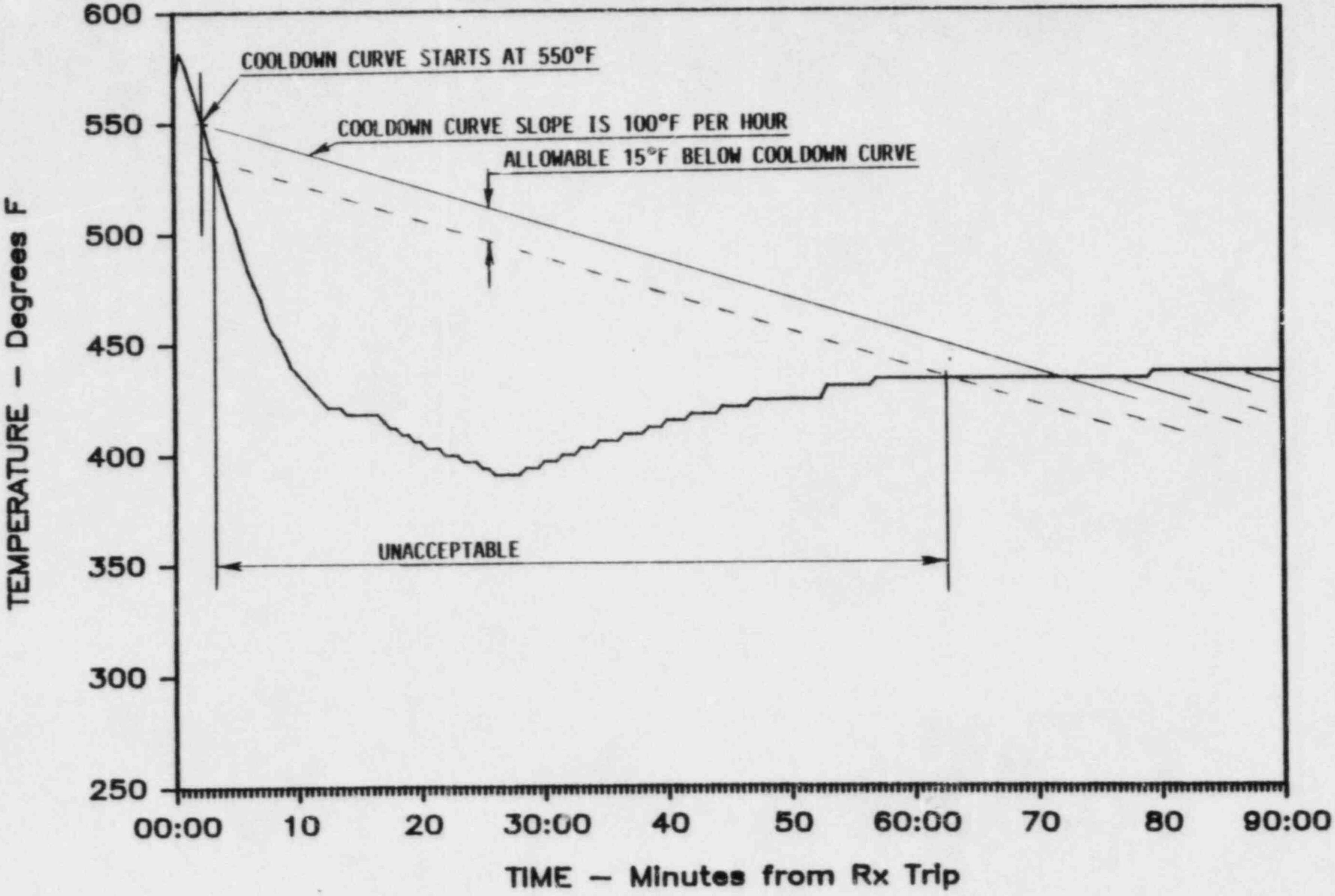
DEC. 5, 1985 Rx TRIP

RCS B–LOOP TEMPERATURE (Ave)



Rx TRIP DEC. 26, 1985

RCS B-LOOP TEMPERATURE (cold)



NEW P/T CURVE GENERATED

- STEP CHANGES ALLOWED
- TEMPERATURE SPECIFIED ABOVE WHICH ANY TRANSIENT ALLOWED
- TECHNICAL SPECIFICATION CHANGE WILL BE REQUESTED

PRESSURIZED THERMAL SHOCK

BOB DIETERICH

SB LOCA ANALYSIS FOLLOWING TMI

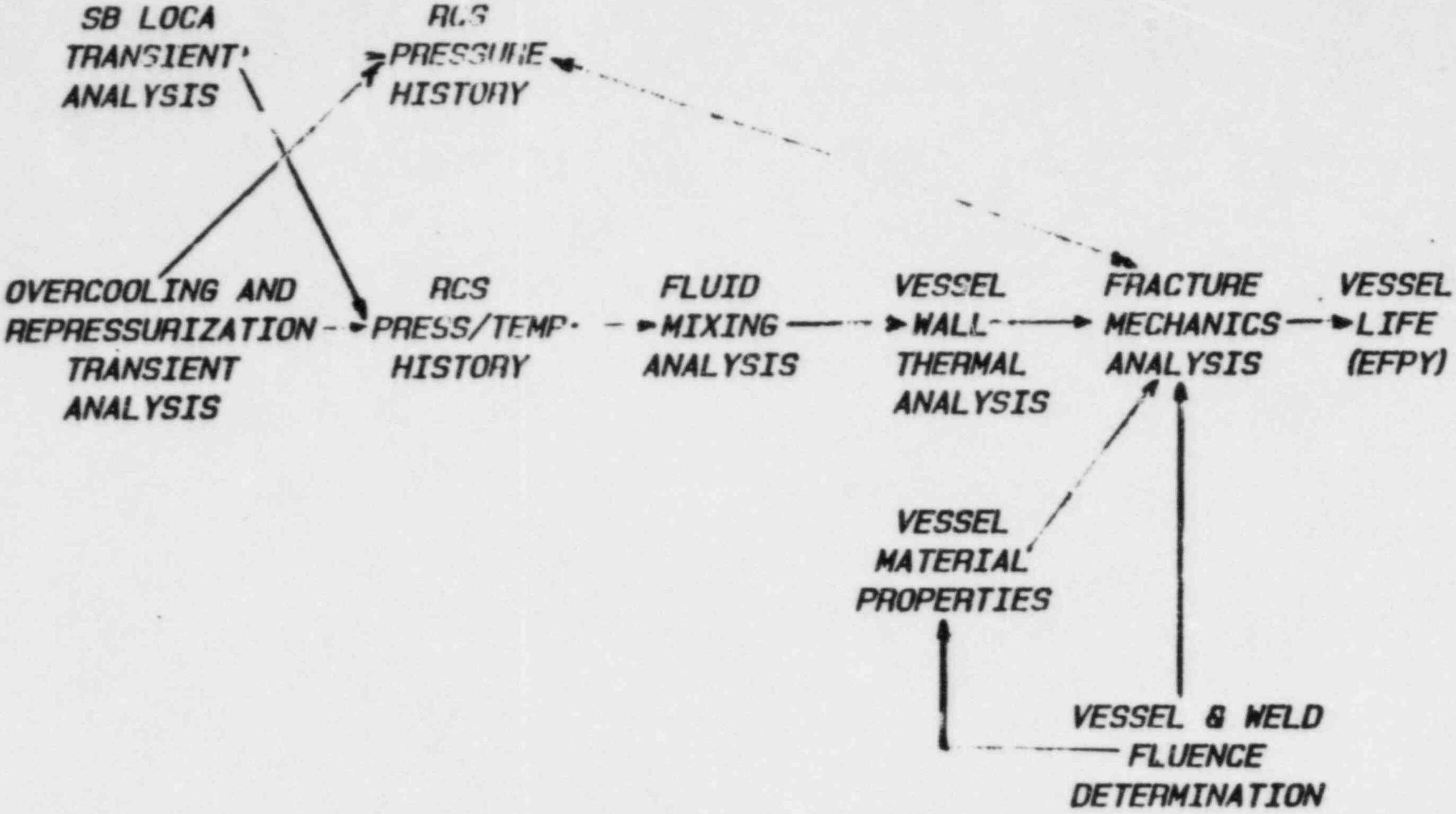
- CORE COOLING
- RANGE OF BREAK SIZES
- HPI ONCE THROUGH COOLING
  FOR LOSS OF ALL FEEDWATER
- WITH "LOOP" COOL HPI WATER
  RAN DOWN VESSEL WALL
- THERMAL SHOCK
- PRESSURIZED THERMAL SHOCK

B & W OWNERS GROUP ANALYSIS

COMPLETED AUGUST 1982

- BAW-1751
- MULTI-DISCIPLINED ANALYSIS

PTS ANALYSIS



SB LOCA TRANSIENT ANALYSIS

RCS PRESSURE HISTORY

OVERCOOLING AND REPRESSURIZATION TRANSIENT ANALYSIS

RCS PRESS/TEMP HISTORY

FLUID MIXING ANALYSIS

VESSEL WALL THERMAL ANALYSIS

FRACTURE MECHANICS ANALYSIS

VESSEL LIFE (EFPY)

VESSEL MATERIAL PROPERTIES

VESSEL & WELD FLUENCE DETERMINATION

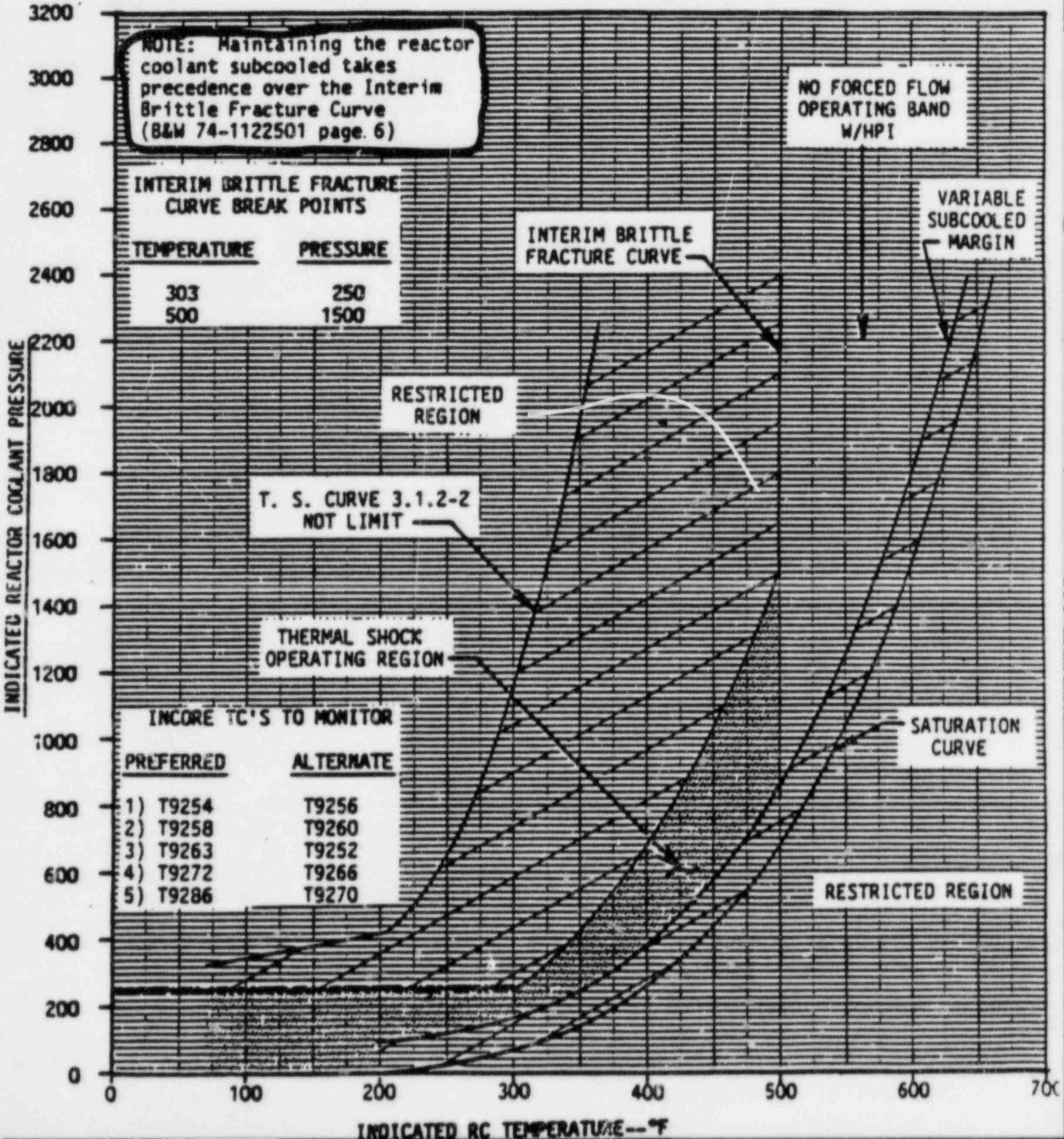
GUIDANCE TO OPERATORS

- BASED ON PTS STUDY
- CONSERVATIVE ASSUMPTIONS
  - NO FORCED FLOW
  - 32EFPY
  - WORST FLAW
  - COOLDOWN OVER 100 F/HR
  - REPRESSURIZATION TO 2500 PSI

ENCLOSURE 8:1

FIGURE 1

REACTOR COOLANT SYSTEM INTERIM FRACTURE LIMIT
NO FORCED FLOW AND THERMAL SHOCK OPERATING REGION



NOTE: Maintaining the reactor coolant subcooled takes precedence over the Interim Brittle Fracture Curve (B&W 74-1122501 page 6)

INTERIM BRITTLE FRACTURE CURVE BREAK POINTS

TEMPERATURE	PRESSURE
303	250
500	1500

INCORE TC'S TO MONITOR

PREFERRED	ALTERNATE
1) T9254	T9256
2) T9258	T9260
3) T9263	T9252
4) T9272	T9266
5) T9286	T9270

START-UP SCHEDULE

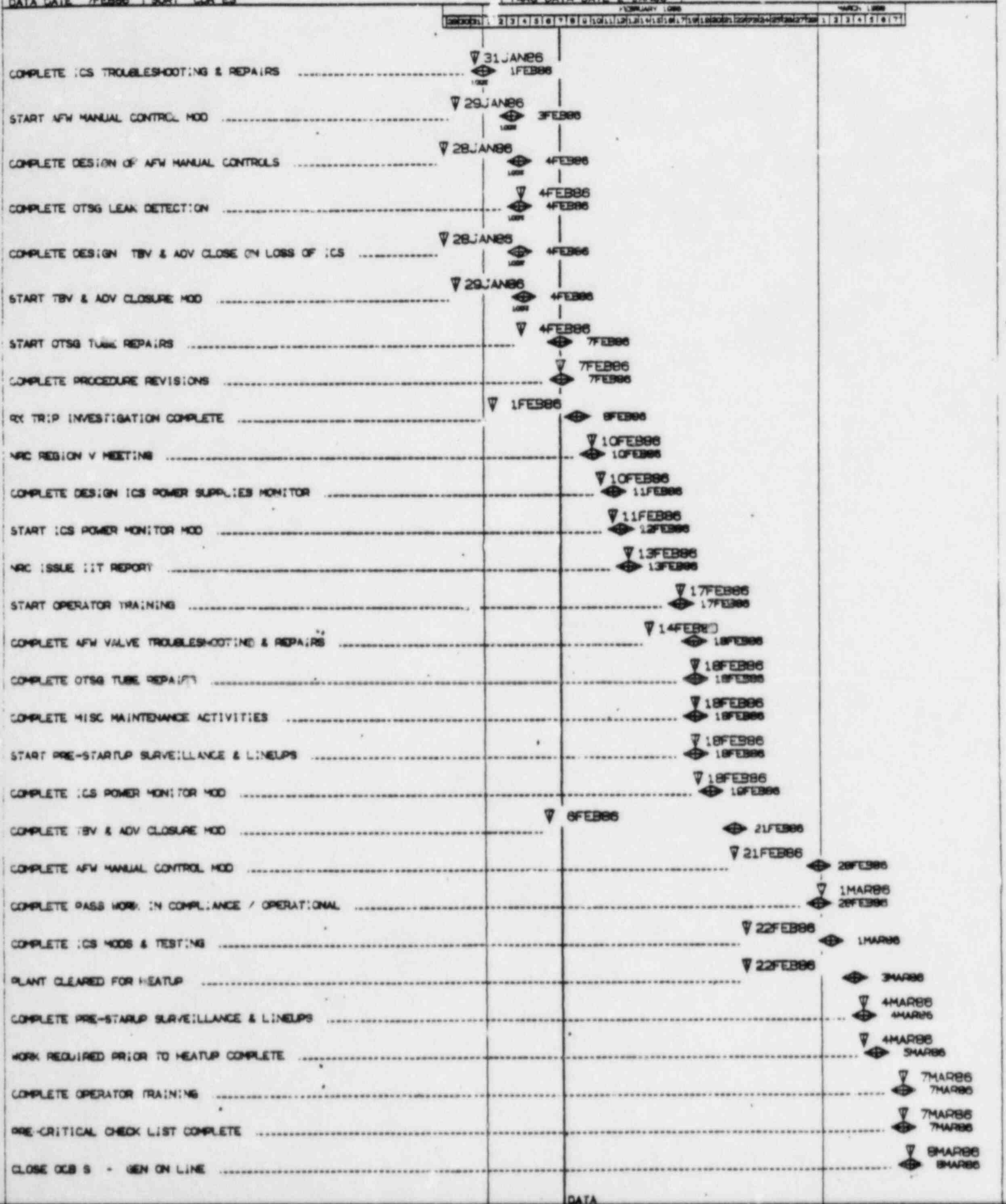
RON RODRIGUEZ

PROJECT MUP
PLOT TAR
PAGE 1 SHEET 1
START 26DEC85
FINISH 8MAR86
DATA DATE 7FEB86

MAJOR MILESTONE SCHEDULE (TAR 3)
RANCHO SECO NUCLEAR GENERATING STA
S M U D NUCLEAR OPERATIONS
TARGET SCHEDULE
SORT CUR ES

MODE T/FE
INTERVAL 24 PERIOD(S)
TARGET SCHEDULE 3
TARG FINISH 8MAR86
TARG DATA DATE 27JAN86

RUN 7FEB86 14:05
PROJECT/2 85A2
SCHEDULE BAR CHART



COMPLETE ICS TROUBLESHOOTING & REPAIRS	31JAN86	1FEB86
START AFW MANUAL CONTROL MOD	29JAN86	3FEB86
COMPLETE DESIGN OF AFW MANUAL CONTROLS	28JAN86	4FEB86
COMPLETE OTSG LEAK DETECTION	4FEB86	4FEB86
COMPLETE DESIGN TBV & ADV CLOSE ON LOSS OF ICS	28JAN86	4FEB86
START TBV & ADV CLOSURE MOD	29JAN86	4FEB86
START OTSG TUBE REPAIRS	4FEB86	7FEB86
COMPLETE PROCEDURE REVISIONS	7FEB86	7FEB86
RX TRIP INVESTIGATION COMPLETE	1FEB86	8FEB86
NRC REGION V MEETING	10FEB86	10FEB86
COMPLETE DESIGN ICS POWER SUPPLIES MONITOR	10FEB86	11FEB86
START ICS POWER MONITOR MOD	11FEB86	12FEB86
NRC ISSUE IIT REPORT	13FEB86	13FEB86
START OPERATOR TRAINING	17FEB86	17FEB86
COMPLETE AFW VALVE TROUBLESHOOTING & REPAIRS	14FEB86	18FEB86
COMPLETE OTSG TUBE REPAIRS	18FEB86	18FEB86
COMPLETE MISC MAINTENANCE ACTIVITIES	18FEB86	18FEB86
START PRE-STARTUP SURVEILLANCE & LINEUPS	18FEB86	18FEB86
COMPLETE ICS POWER MONITOR MOD	18FEB86	19FEB86
COMPLETE TBV & ADV CLOSURE MOD	6FEB86	21FEB86
COMPLETE AFW MANUAL CONTROL MOD	21FEB86	28FEB86
COMPLETE PASS WORK IN COMPLIANCE / OPERATIONAL	1MAR86	28FEB86
COMPLETE ICS MODS & TESTING	22FEB86	1MAR86
PLANT CLEARED FOR HEATUP	22FEB86	3MAR86
COMPLETE PRE-STARUP SURVEILLANCE & LINEUPS	4MAR86	4MAR86
WORK REQUIRED PRIOR TO HEATUP COMPLETE	4MAR86	5MAR86
COMPLETE OPERATOR TRAINING	7MAR86	7MAR86
PRE-CRITICAL CHECK LIST COMPLETE	7MAR86	7MAR86
CLOSE OCB S - GEN ON LINE	8MAR86	8MAR86

Attachment 2

OPERATION

LICENSEE ACTIONS

1. Develop procedures for monitoring conditions within the AFWS and for recognizing steam binding and restoring the AFWS to operability (CCL No. 850842) (Arises from NRC Bulletin No. 85-01).

2. Review procedures to prevent water in main steam lines.

3. Develop procedures for swapping from AFW to MFW (Recommendation #12 from Trip Report #73). (CCL No. 860019).

4. Evaluate need for event related procedures in addition to symptom related procedures (derived from the ATOGs).

5. Propose amendment that will revise the Technical Specifications to clarify the applicability of the reactor vessel maximum allowable cooldown rate.

6. Provide technical basis for pressurized thermal shock guidelines.

7. Provide clearer operator guidance for pressurized thermal shock concerns (Recommendation #15 from 10/2/85 Trip Report #73). (CCL No. 860022).

8 Perform loss of ICS procedure evaluations. Review against ATOG.

9 Conduct training on overcooling events (Recommendation #21 from 10/2/85 Trip Report #73). (CCL 860028).

10. Evaluate adequacy of operator training and the need for operator reevaluation and retraining, including specific training on ICS off normal operation, EPIPs CVCS, PTS recovery action, HPI and AFW throttling and trip criteria, contaminated area entry, communications, between operations personnel and other personnel, ADV operation, and differences between the B&W simulator and Rancho Seco plant.

11. Evaluate the significance of the incapacitated senior operator in terms of his fitness for duty.

OPERATION

LICENSEE ACTIONS

12. Evaluate training on Emergency Operating Procedures (Recommendation #22 from 10/2/85 Trip Report #73). (CCL No. 860029)

13. Evaluate need to correct noise problem with control room HVAC.

14. Evaluate adequacy of valve position indication for TBVs, ADVs and AFW flow control valves.

ENGINEERIN

LICENSEE ACTIONS

1 Perform walkdown of the secondary steam system to determine branch lines that have potential for contributing to an overcooling event. For those lines whose valves do not isolate on loss of ICS, ensure that the valve motor operators are capable of control from the control room independent of the ICS.

2. Assure the availability of uninterrupted water supply whenever makeup/HPI pumps are operating by determining the root cause for makeup pump damage and evaluating the need for valve interlocks for various sources of water supply to operating makeup/HPI pumps.

3. Perform Root Cause Analysis for 12/26 event (2.a).

4, For the loss of DC power to ICS on 12/26, determine the immediate cause and provide corrective action to resolve any single failure identified.

5, Licensee to perform trouble-shooting of ICS (3.b & 3.c).

6, Provide analysis of the thermal transient effects on the reactor vessel and steam generators following the Dec. 26 loss of ICS power.

ENGINEERING

LICENSEE ACTIONS

7. Complete Post Trip Report (1)

8. Perform Analysis of Steam Lines to include stress calcuations and walkdown inspections (1.f)

9. Determine minimum level reached by pressurizer and potential for RV head void (1.g)

10. Test condition of pressurizer heaters (1.i)

11. Determine and identify CR instruments which fail on loss of ICS power (1.j)

12. Investigate MF block valve operation in main steam failure logic as shown by IDADS (1.n)

13. Examine operator reference to strip charts vs. SPDS for OTSG level (1.0)

14. Determine if 12/26 transient was within USAR design basis (1.p)

15. Determine cause of MU pump (p. 236) failure (4.b)

16. Licensee to modify procedures to permit operation without Make Up Pump (4.c)

LICENSEE ACTIONS

17 Licensee to provide schedule for repair/replacement of MU pump.

18. Prepare LER on RCS overcooling (6.a)

19. B&W to analyze potential for core lift during 12/26 transient (6.b.1)

20. Evaluate acceptability of closing maintenance valves during a transient.

MAINTENANCE

LICENSEE ACTIONS

1. [redacted] Implement a valve preventative maintenance program (11.e.h) (Not restricted to safety related valves).

2, [redacted]. Revise I&C and electrical preventative maintenance procedures to include a step to check termination tightness. (CCL Item No. 850744 and 850745)

3. [redacted]. Failure of "B" control rod drive breaker to trip (NRC open item 82-19-10 with follow-up IR 82-36 possible generic item).

LICENSEE ACTIONS

4, Verify operability of manual valves and manual operators of control valves. Perform inspection of valve packing to assure integrity and proper assembly.

5. Perform repairs to ICS and valves which were damaged during the 12/26 transient.

MODIFICATIONS

LICENSEE ACTIONS

1. Examine alternatives to ADVs and TBVs opening to midposition on loss of ICS. (3.f.4)

2. Examine present response of MFW valve on loss of ICS and response of MFW pump speed control on loss of ICS.

3. Establish the capability for control room control, independent of the ICS, of AFW flow control valves, ADVs and TBVs. (3.f.5)

4. Evaluate the effects on the system following the restoration of ICS power and determine what modificatations, if any, that should be made.

5. Evaluate use of multi-channel ICS/NNI power supply monitors.

6. Evaluate "ICS trouble" annunciator response procedure.

7. Evaluate the use of an interlock between MU tank discharge valve and BWST valve.

8. Examine IN 85-94 for applicability (4.h)

RADIATION PROTECTION &

LICENSEE ACTIONS

1 Improve communications between operations and chem./rad. protection so supervisors are aware of changes in plant conditions. ~~(LRS Study Item CRP-9)~~

2, Establish a PASS Program.

3. Provide a definitive schedule for resolving HP/Chem. issues. (NRC, LRS, EB findings)

4. Establish a revised schedule for the long range control of liquid effluent releases.

5. Revise EPIP's to provide current setpoints for effluent releases.

6. The licensee should revise AP 509 to allow th~~e on-shift crew to~~ project off-site dose resulting from a steam generator tube rupture.

SECURITY ISSUES

LICENSEE ACTIONS

1. Evaluate the effect of security on operator's ability to gain access and control of critical equipment in an emergency. This is to include:
   - a. Review of procedures
   - b. Security response to badge loss.
   - c. Locked door control policy & procedures.

Attachment 3
SM

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

February 5, 1986

Docket No. 50-312

MEMORANDUM FOR: Harold Denton, Director
Office of Nuclear Reactor Regulation

FROM: Frank Miraglia, Director
Division of PWR Licensing-B

SUBJECT: ACTION PLAN RECOMMENDATIONS IN RESPONSE TO DECEMBER 26, 1985 RANCHO SECO OVERCOOLING EVENT

The purpose of this memorandum is to forward, for your approval, the staff's proposed plan for corrective action (Enclosure 1) in response to the December 26, 1985 overcooling event at the Rancho Seco Nuclear Generating Station. This action plan is the result of our preliminary assessment of the sequence of events which occurred during the December 26 transient and reflects the need for corrective action prior to plant restart. This action plan is intended to provide the licensee with our early assessment of the problems and deficiencies identified in our review of the event and the corresponding issues which need to be resolved. While we are well aware that the EDO's Incident Investigation Team (IIT) is presently conducting an in-depth investigation of the incident to develop the bases for subsequent corrective actions, it is our intention to provide the licensee with the enclosed preliminary action plan, well in advance of any actions which may develop from the IIT findings and subsequent EDO direction. (The IIT report is expected to be issued in mid-February). Independent of our efforts, it should be noted that the licensee has developed and implemented a plan for corrective action and provided us a status of their ongoing effort (See Enclosure 2, letter dated January 24, 1986 from R. Rodriguez to J. Martin and F. Miraglia). The licensee's January 24 letter stated that the District is working towards a restart schedule of March 8, 1986. Regardless of whether this schedule is realistic or not, we think it's very important that our action plan be integrated into the licensee's as soon as possible as the most efficient and effective means of resolving identified issues. We note that, in comparing the licensee's action plan with ours, there are issues we've identified which are not embodied in the licensee's plan but which require resolution. In this regard, a meeting has been scheduled in Region V on February 10 to discuss the status of the licensee's efforts and this meeting could be a useful forum for integrating our action plan into the licensee's.

Our action plan was developed in conjunction with Region V and coordinated with the Office of Inspection and Enforcement staff. The plan also reflects comments provided by members of the IIT. As such, the plan represents the collective views of the staff. The plan identifies the staff's concerns in the following five areas as well as the issues which must be resolved, some prior to plant restart:

1. Plant mechanical, electrical and control systems deficiencies
2. Adequacy of plant maintenance
3. Adequacy of operator training
4. Adequacy of plant normal and emergency procedures
5. Human engineering deficiencies

While the plan focuses on issues specific to Rancho Seco, there may well be generic implications for other B&W plants. However, based on information presented at the January 8, 1986 meeting with the B&W Regulatory Response Group, the staff has concluded that no immediate action is required for the other B&W plants with respect to the Rancho Seco event. The generic actions which develop from this effort will be addressed by separate memorandum. Further, as a result of this event and other recent events at B&W designed reactors, we are presently developing a detailed program plan to evaluate the basic design requirements for B&W reactors. Our schedule is to develop the plan by mid-February and to complete the evaluation this calendar year. While we believe that this evaluation is needed, we also believe that B&W reactors can safely continue to operate in the interim.

For the reasons stated above, I am recommending that the enclosed action plan be forwarded to the District so that a coordinated effort can be focused on resolution of identified issues. I am available to discuss this plan at your convenience.

Frank J. Miraglia

Frank Miraglia, Director
Division of PWR Licensing-B

Enclosures:
As Stated

cc w/enclosures:
J. Martin, RV
J. Taylor, I&E

Enclosure

Action Plan for Resolution of Issues

Listed below are the specific issues in the five areas of concern which we intend to resolve as the appropriate staff response to the Rancho Seco overcooling event. Those items which will require resolution prior to restart are specifically identified.

Responsibility I. Plant Mechanical, Electrical and Control Systems Deficiencies

NRR
PWR-B 1. Loss of DC power to ICS.

Action: a. *Determine root cause and provide corrective action to resolve single failure identified.

b. *Identify the control room instruments which fail on loss of ICS power and provide an alternate power supply for these instruments.

c. Provide failure modes and effects analysis (FMEA) of ICS with interfacing systems to determine single failure points and their effects for loss of ICS power. A six month timetable seems appropriate.

NRR
PWR-B 2. Plant control on loss of ICS resulting in significant overcooling event. Evaluate the design and procedural changes necessary to minimize transients induced by loss of ICS. Design considerations should include, as a minimum, the items cited in A, B, and C.

A. Steam losses through ADVs and TBVs.

Action: a. *Evaluate alternatives to ADVs and TBVs opening to midposition on loss of ICS.

B. Valve position changes and reduction of MFW pump speed on loss of ICS.

Action: a. *Evaluate alternatives to present design response of (1) startup and MFW control valves closing to midposition and (2) reduction of MFW pump speed to 2500 rpm on loss of ICS.

C. Control of AFW flow control valves, ADVs and TBVs independent of ICS.

Action: a. *Provide capability for control room control, independent of the ICS, of AFW flow control valves, ADVs and TBVs. The effects of restoration of ICS power should, likewise, be evaluated.

Region V (except NRR to provide criteria) b. *Perform walkdown of the secondary steam system to determine branch lines that have potential for contributing to an overcooling event. For those lines whose valves do not isolate on loss of ICS, ensure that the valve motor operators are capable of control from the control room independent of the ICS.

*These items must be completed prior to plant restart.

Responsibility

NRR
PWR-B
3. Plant modifications made to B&W plants which minimize effects of ICS induced transients.

All other B&W plants with the same ICS control system have made modifications to the plant so that the results of ICS failure transients are significantly less severe.

Action:
a. *Explain why SMUD has not installed similar modifications to date, e.g., EFIC.

b. Does SMUD keep current with modifications made at other facilities to improve plant safety and evaluate whether the same modifications would provide similar enhancement at Rancho Seco?

Reg Guide 1.97

c. Are there other areas where SMUD has not implemented improvements made at other B&W facilities which would improve safe operation at Rancho Seco.

NRR
PWR-B
4. Availability of uninterrupted water supply whenever makeup/HPI pumps are operating.

Action:
a. *Determine root cause for makeup pump damage.

b. *Evaluate need for valve interlocks for various sources of water supply to operating makeup/HPI pumps.

NRR
PWR-B
5. Effects of overcooling event on the reactor vessel and steam generators and limiting cooldown rate in the Technical Specifications.

Action:
a. *Provide analysis of the thermal transient effects on the reactor vessel and steam generators

b. *Provide amendment that will revise the Technical Specifications to clarify the applicability of the reactor vessel maximum allowable cooldown rate.

c. *Provide technical basis for pressurized thermal shock guidelines.

NRR
PWR-B
6. Radiation monitor system damage after ESFAS actuation.

Action:
a. Determine root cause for radiation monitor system damage.

b. Review the consequences of containment isolation on systems that are required to operate following ESFAS actuation (e.g., containment radiation monitor).

Region V
7. Flooding of the main steam headers.

Action:
a. *Evaluate main steam header supports to assure that no damage occurred when overfill of steam generator allowed water to enter the main steam headers.

Responsibility II. Adequacy of Plant Maintenance, Testing and Surveillance

This event revealed valve performance shortcomings which may have been partially caused by insufficient valve maintenance. The 1984 and 1985 SALPs recommended that SMUD adopt a valve maintenance program. Primary coolant system leaks and decay heat removal system leaks in December, 1985 also point to the possibility of excessive deferred valve maintenance. Other maintenance program weaknesses are also well documented.

Region V Action: a. *Evaluate the Rancho Seco maintenance program, and the SMUD schedule and program to identify and correct known weaknesses. This evaluation should include:

1. Manual valve operability;
2. Control valve operator physical integrity and operability;
3. Integrity and proper assembly of valve packing;
4. Other previously identified maintenance program deficiencies to ensure that timely action to correct these deficiencies has occurred, or is occurring. Sources for these deficiencies include SALP, inspection reports, and NRC - licensee meetings.

Responsibility III. Adequacy of Operator Performance

1. Operator deficiencies cited in response to December 26, 1985 overcooling event and in other events during past several months.

Region V Action: a. *Evaluate adequacy of licensed and non licensed operator training and the need for operator reevaluation and retraining, including specific training on ICS off normal operation, CVCS, PTS recovery action, HPI and AFW throttling and trip criteria, contaminated area entry, communications between operations personnel and other personnel, ADV operation, and differences between the B&W simulator and Rancho Seco plant.

NRR b. Evaluate minimum staffing requirements for Rancho Seco.

Region V c. *Evaluate the significance of the incapacitated senior operator in terms of his fitness for duty.

Region V d. *Conduct a safeguards team inspection to determine the extent of potential security/safety interface issues, if any, and identify any requiring timely corrective action by the licensee.

Responsibility IV. Adequacy of Plant Normal and Emergency Procedures

1. Procedural deficiencies noted in response to December 26, 1985 overcooling event and in earlier events.

NRR
PWR-B — Action: a. *Evaluate need for event related procedures in addition to symptom related procedures (derived from the ATOGs).

NRR
PWR-B — b. *Evaluate the adequacy of the ATOG specifically regarding the impact of continuing feedwater flow which results in overcooling that exceeds PTS procedural guidelines, continuing high pressure injection once plant begins to repressurize, heating the plant up following a rapid cooldown, condensate pump injection, etc.

Region V — c. *Evaluate implementation of health physics procedures during the emergency.

Region V — d. *Evaluate the consistency, if any, of the Rancho Seco procedures with the ATOG. The ATOG at the time of the event as well as after any enhancements derived from this event should be used as source references.

Responsibility V. Human Engineering Deficiencies

Region V — 1. Human engineering deficiencies exacerbated licensee response to December 26, 1985 overcooling event.

Action: a. *Evaluate need to better identify breaker position in ICS cabinets.

b.*Evaluate adequacy of valve position indication for TBVs, ADVs and AFW flow control valves.

c.*Evaluate need to correct noise problem with control room HVAC. (NRR to supply criteria).