LP&L LOUISIANA POWER & LIGHT / 142 DELARONDE STREET P. O. BOX 6008 · NEW ORLEANS, LOUISIANA 70174 · (504) 366-2345
MIDDLE SOUTH UTILITIES SYSTEM

October 21, 1985

W3P85-3244
A4.05
QA

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

Dear Mr. Knighton:

Subject: Waterford 3 SES
Docket No. 50-382
License No. NPF-38
OPERATING SHIFT STAFFING

This letter provides some updated information on the Operating Shift Staffing for Waterford 3. The information is submitted as agreed during the recent telephone conversation between personnel of the LP&L Training Department and NRC Staff. Some of the attachments contain personal information and it is requested that they be withheld from public disclosure.

The following information in a question/response format covers the matters discussed by telephone.

Question 1

How are your shifts currently covered with respect to having an experienced person on each shift?

Response 1

Waterford 3 is currently operating on 5-shift rotation. Each shift has an experienced operator who has a Waterford 3 Senior Reactor Operator license.

Question 2

Provide details on the experience of your Shift Advisors and their training program.

Response 2

Attachments 1 and 2 are resume's for Waterford 3 Shift Advisors, Mark K. Phillippe and David L. Shipman. These attachments contain personal information and we kindly request that these attachments be withheld from public disclosure pursuant to 10CFR2.790(a)(6).

Change: 24 x LPDR PDR NSIC } Ltr Only

M003

1/1

Attachment 3 provides the detail on their training program.

Question 3

Provide a synopsis of the operating history of the plant since fuel load in December 1984.

Response 3

A power history and a description of plant operations is provided in Attachment 4.

We plan to use all time above Mode 5 (Modes 1-4) as time that counts towards meeting the six months on shift portion of the hot participation experience requirements. As of October 16, 1985 we had approximately 160 cumulative days of Mode 4 operations or above.

Should you have any questions regarding this matter, please call me at (504) 595-2805 or Lew Myers, Operations Superintendent, at (504) 464-3118.

Very truly yours,

K.W. Cook
Nuclear Support & Licensing Manager

KWC:GEW:sms

Enclosures

cc: R.D. Martin, NRC Region IV
R.A. Cooley, NRC Region IV
J.H. Wilson, NRC-NRR
L.P. Crocker NRC (LQB)
NRC Resident Inspectors Office
B.W. Churchill
W.M. Stevenson

ATTACHMENT 3

TRANSCRIBED FROM ORIGINAL WEEKLY SCHEDULES

W-3 MON

REVIEWED: David P. Clark / 10/18/85
Date

APPROVED: [signature] / 10/18/85
Date

SUNDAY	MONDAY	TUESD
W-1	5 INTRODUCTION TO TECH SPECS REACTOR COOLANT SYSTEM	6 REACTOR COO SYSTEM (Co REACTOR VES INTERN
W-2	12 Electrical Distribution Overview Main Generator 6.9 KV 4.16 KV Emergency Diesel Generator	13 Electrical Dis 480 VAC Low voltage Batteries Inverters
W-3	19 Reactor Theory	20 Reactor
W-4	26 Shutdown Cooling & Low Pressure Safety Injection High Pressure Safety Injection Systems LOCA (Seminar)	27 Ventilation S (Reactor Auxil HVAC/Contain Isolation/Con Cooling & Ven Shield Bldg Ve MSLB Inside C
W-5	2 HOLIDAY	3 Excore Nuclear Instrumentation Reactor Coolant Instrumentation Reactor Protecti

THLY TRAINING SCHEDULE

SHIFT ADVISOR TRAINING
AUGUST - SEPTEMBER 1985

DISTRIBUTION
Asst. Plant Mgr. O.M.
Asst. Plant Mgr. T.S
Plant Trng. Mgr.
Ops. Supt.
STA Supt.
Ops. Trng. Dept. (9)
Control Room (6)

AY	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
ANT ... ... CL ... ALS	7 Control Room HVAC SECONDARY SYSTEMS OVERVIEW (Main, Auxiliary & Gland Steam/Condensate/Auxiliary and Main Feedwater/Heaters Vents & Drains)	8 Secondary Systems (Cont) (Extraction Steam/Circ Water/Air Evacuation/Instrument Air/Turbine & Turbine Control/Turbine Support Systems)	9 Secondary Systems (Cont) (Turbine Bldg Cooling Water/Steam Generator Blowdown/Water Treatment) Plant Monitoring Computer	
...ibution ...stribution ... ...stribution	14 Heat Transfer & Fluid Flow	15 Reactor Thermal Hydraulics	16 Reactor Thermal Hydraulics Quiz	
Theory	21 Reactor Theory & Core Parameters	22 Chemical & Volume Control System Boric Acid Makeup Pressurizer Level & Pressure Control	23 Fuel Handling & Storage Emergency Feedwater System	
...stems ...ary Bldg ...t & ...ainment ...ilation/ ...tilation ...t (SEMINAR)	28 Containment Spray Component Cooling and Auxiliary Component Cooling Water System Post Accident Long-Term Cooling (Seminar)	29 Plant Protection System Overview Engineered Safety Features Actuation Systems Essential Chilled Water	30 Quiz	
... Instru... ... System ...tion ...on System	4 Reactor Protection System (continued) 10 CFR 50.46 Criteria Core Protection Calculators	5 Control Systems Steam Bypass Control System Control Element Drive Mechanism Control System Reactor Regulating System Feedwater Control System Reactor Power Cutback System Vibration & Loose Parts Monitor & Seismic Monitor	6 Incore Nuclear Instr. Core Operating Limit Supervisory System Azimuthal Tilt & RCS Flowrate Calculation with COLSS Inoperable	

8510280329-01

TRANSCRIBED FROM ORIGINAL WEEKLY SCHEDULES.

W-3 MONT

REVIEWED: David P. Clark 10/18/85
Date

APPROVED: [illegible] 10/18/85
Date

SUNDAY	MONDAY	TUESDAY
W-6	9 Mitigating Core Damage • Incore Instrumentation • Excore Instrumentation	10 Mitigating Core Damage • Radiochemistry • Radiation Monitors • Gas Generation
W-7	16 Normal Operating Procedures (OP10-001) Administrative Procedures (including Event Evaluation & Reporting (AT-6-010)	17 Selected Off-normal Procedures Function Based Emergency Operating Procedures

DISTRIBUTION
Asst. Plant Mgr. O.M.
Asst. Plant Mgr. T.S
Plant Trng. Mgr.
Ops. Supt.
STA Supt.
Ops. Trng. Dept. (9)
Control Room (6)

THLY TRAINING SCHEDULE

SHIFT ADVISOR TRAINING
SEPTEMBER - 1985

Y	WEDNESDAY	THURSDAY	FRIDAY	SATURDAY
e ... ing	11 INDUSTRY CONCERN & SIGNIFICANT EVENTS	12 SAFETY PARAMETER DISPLAY SYSTEM & QUALIFIED SAFETY PARAMETER DISPLAY SYSTEM	13 QUIZ WASTE MANAGEMENT (LIQUID/BORON/GAS/SOLID)	
nal ... ating	18 Function Based Emergency Operating Procedures	19 Contract & Review	20 SHIFT ADVISOR CERTIFICATION EXAMINATION	

8510280329-02

ATTACHMENT 5

COURSE NAME Shift Advisor　　　COURSE NUMBER R985-000-50

EXAM TITLE Certification Exam　　　EXAM DATE 9/20/85

TIME LIMIT: 3 Hrs. TIME START: ______ TIME FINISH: ______ CATEGORY I (II) (circle one)

MINIMUM ACCEPTABLE GRADES . . . . . EACH SECTION: 70 % . . . . . OVERALL: 80 %

NAME: ______________________ SOCIAL SECURITY # ______________
Last　　First　　MI

****

INSTRUCTIONS AND GUIDELINES

PLEASE READ THE FOLLOWING INSTRUCTIONS CAREFULLY:

1) Use only black ink or pencil (No. 2 or softer).
2) If you have any questions during the examination, please raise your hand. Your instructor will provide clarification wherever possible.
3) You are expected to do your own work and you are not to help anyone else.
4) Use only the reference material provided.
5) After completion of this examination, you are to sign the following certification:

I certify all answers contained in this examination are my own. In addition, I have not received nor given any unauthorized assistance, nor have I used any unauthorized references.

SIGNATURE: ______________________ DATE: ______________

****

Category	Category Value	% of Total	Applicant's Score	% of Cat. Value
1) Plant Systems and Designs for Mitigating Effects of Core Damage.	17	51		
2) Plant Procedures and Controls and Limitations.	16.5	49		
TOTALS/OVERALL/FINAL GRADE	33.5	100		

****

AUTHORIZED REFERENCE MATERIALS: None

SPECIAL INSTRUCTIONS: None

****

Prepared by: C R Gaines (Examiner)　　Date: 9/20/85

Reviewed by: David P. Clark (Training Supervisor)　　Date: 9/20/85

Approved by: [signature] (Training Superintendent)　　Date: 9/20/85

1. PLANT SYSTEMS AND DESIGNS FOR MITIGATING EFFECTS OF CORE DAMAGE

(1.0) 1.1 List the Emergency Core Cooling systems acceptance criteria required by 10CFR50.

(3.0) 1.2 Sketch a simplified diagram of the Electrical Distribution System from the Main Generator and offsite up to and including all 4.16 KV Buses. Include all sources of power and label all buses and major components.

(1.0) 1.3 List the four major contributors of hydrogen in the containment in a LOCA environment.

(0.5) 1.4 A small break Loss of Coolant Accident occurred forty-five (45) minutes ago. The Primary NPO notes that the Startup Channel recorders are trending up. They have risen 1½ decades over the last 25 minutes.

Briefly explain the significance of (reason for) this indication.

(1.0) 1.5 List three (3) basic design barriers against release of radioactive material offsite from the core.

(1.5) 1.6 With regard to CPC's, answer the following:

a) List the inputs that CPC's use to calculate the Departure from Nucleate Boiling Ratio value.

b) List the CPC Auxiliary Trips (setpoints not required).

c) What indications would you receive if a CPC auxiliary trip occurred?

(1.0) 1.7 Elevated Containment temperature cause erroneous Pressurizer Level indication due to __________________. Indicated level will be ____________ than actual level because of ________________________ seen by the level transmitter.

(0.5) 1.8 What are the two consequences associated with having a hydrogen burn or a hydrogen explosion inside containment?

(0.5) 1.9 List the actuation signal and setpoint for recirculation actuation signal.

(2.0) 1.10 What is DNB? Why is avoided in a reactor?

(2.0) 1.11 The core is operating in the nucleate boiling region. Reactor Coolant pressure is increased. What effect does this have on heat transfer at the clad/coolant interface.

Explain your answer.

(3.0) 1.12 Regarding the ESFAS System what conditions (parameters and setpoints) will initiate the following:

a. SIAS
b. CSAS
c. CIAS
d. MSIS

- END OF SECTION 1 -

2. PROCEDURES - NORMAL, OFF-NORMAL, EMERGENCY, ADMINISTRATIVE AND CONTROLS AND LIMITATIONS

(2.0) 2.1 List all conditions which require Emergency Boration per OP-901-013 - Off-Normal Procedure "Emergency Boration".

(0.5) 2.2 A Steam Generator Tube Leak occurs and the operators are carrying out the required actions of OP-901-024 "Steam Generator Tube Leakage or High Activity". You observe conditions that classify the leak to actually be a "Steam Generator Tube Rupture". What conditions do you observe to make this determination?

(0.5) 2.3 Choose, from the list below, the symptom that would most clearly differentiate between a large break LOCA and a large Main Steam Line Break inside containment:

a) Increasing Safety Injection Sump Level

b) Rapidly Decreasing Pressurizer Level

c) Increasing Containment Radiation Levels

d) Rapidly Decreasing Tavg

(1.0) 2.4 Under what condition(s) is the Emergency Entry Procedure implemented?

(1.5) 2.5 List the three (3) general occurences or conditions resulting in implementation of OP-902-008 - Safety Function Recovery Procedure.

(1.0) 2.6 One of the immediate actions of OP-902-000 - Emergency Entry Procedure is to "Check Feedwater Control in Reactor Trip Override".

a) What happens on a Reactor Trip Override?

b) What is the purpose of Reactor Trip Override?

c) What would intentionally cause any of the actions in your answer to part "a" to not occur?

(1.0) 2.7 While you are assigned as the On-Duty Shift Advisor, a Loss of Main Feedwater occurs from operation at 80% power:

What are the plant conditions (including setpoints and logics) which would initiate EFAS-1 and EFAS-1 signals?

(1.5) 2.8 Fill in the blanks in the following three (3) questions relating to the Safety Limits at Waterford-3:

a) The __________ of the reactor core shall be maintained greater than or equal to ____________.

b) The ________________ of the fuel shall be maintained less than or equal to ___________.

c) The __________ pressure shall not exceed _____________ psia.

(2.0) 2.9 Answer the following questions regarding SHUTDOWN MARGIN:

a) Define SHUTDOWN MARGIN as per Technical Specifications.

b) What are the SHUTDOWN MARGIN requirements for all plant modes <u>other than Refueling</u>?

c) Explain why the required SHUTDOWN MARGIN in your answer to part "b" are dependent on the plant mode? (i.e., Why not one (1) value for all modes?)

d) SHUTDOWN MARGIN is not manually calculated in MODE 1, yet the Surveillance Requirement is to determine it to be within its limits at least once every 12 hours.

How is SHUTDOWN MARGIN determined in this MODE?

(1.0) 2.10 Answer the following TRUE or FALSE:

a) Where immediate action is required to prevent or mitigate the consequences of unforeseen emergency conditions, such action may be taken and later documented as a procedure change.

b) If the plant is operating at 100% power and the On-Duty STA possesses an SRO license on Waterford-3, the Shift Supervisor can designate him to assume the Control Room command function.

c) The primary NPO notices parameters that indicate a Reactor Trip should have occurred. He must request permission to trip the unit.

d) The Shift Advisor shall read and initial all Control Room Logs for all shifts since the last day he served as Shift Advisor or one (1) week, whichever time frame is shorter.

e) Neither the Shift Supervisor not the On-Duty STA may enter the Containment.

(1.5) 2.11 What are the requirements of a Temporary Approval of a change (deviation) to a POM Procedure?

(2.0) 2.12 List the positions, number of individuals and licenses applicable to satisfy the minimum shift crew composition for all modes.

(1.0) 2.13 Match the symptom with the applicable procedure to be followed immediately after the operator identifies the symptom:

Symptoms

S1 - Uncontrolled CEA Withdrawal

S2 - During auto makeup to the VCT, the selected BAM pump fails to start.

S3 - Plant Computer total failure (Mode 1)

S4 - Engineered Safety Features Actuation Signal (SIAS) present on 2 channels

Procedures

P1 - Inadvertent Positive Reactivity Addition

P2 - Emergency Entry Procedure

P3 - Uncomplicated Reactor Trip Recovery

P4 - CEA or CEDMCS Malfunction

P5 - COLSS inoperable

P6 - Loss of Coolant Accident

P7 - Safety Function Recovery

NA - None of the above procedures apply

S1 ____________ S3 ____________

S2 ____________ S4 ____________

- END OF SECTION 2 -

- END OF EXAM -

Shift Advisor Certification Exam

Answer Key

1. PLANT SYSTEMS AND DESIGNS FOR MITIGATING EFFECTS OF CORE DAMAGE

(1.0) 1.1 Peak Cladding Temperature (0.15) LTE 2200°F (0.15)

Maximum Cladding Oxidation (0.1) LTE 17% of the total cladding thickness (0.1)

Maximum H_2 generation (0.1) LTE 1% of the H_2 generated if all cladding reacted (0.1)

Long Term Cooling (0.1) - Decay Heat Removal (0.1)

Maintain a coolable geometry (0.1)

(3.0) 1.2 SEE ATTACHED DRAWING

(1.0) 1.3 1) Zr - $H_2$0 reaction of clad (0.25)

2) Radiolysis of H_2O (0.25)

3) Corrosion of Zinc structural material (0.25)

4) Corrosion of Aluminium structural material (0.25)

(0.5) 1.4 Indicates increasing void formation (or reduced density) in the core and downcomer.

(1.0) 1.5 (0.33 each)

Fuel Cladding
RCS Pressure Boundary
Containment

(1.5) 1.6 a) (0.05 each)

CEAC Penalty Factors
Radial Peaking Factor
Axial Power Distribution
Power
Temperature
Pressure
Flow

b) Out of operating space (0.1) for Tc (0.1), Pressure (0.1), Radial Peaking Factors (0.1), and ASI (0.1), Quality Margin (Saturation) (0.1), LT 2 RCP's (0.1), CPC Failure (0.1).

c) Low DNBR (0.1) and High LPD trips (0.1) with no pretrips (0.15).

(1.0) 1.7 (0.33 each)

a) Reference Leg Heatup

b) Higher

c) Decrease in Differential Pressure or Density Decrease in Reference Leg

(0.5) 1.8 (0.25 each)

1) Challenges containment integrity.

2) Damage to equipment.

(0.5) 1.9 RWSP Level Low (0.25)

10% (0.25)

(1.0) 1.10 DNB - Departure From Nucleate Boiling (0.5)

At DNB, too much steam is produced on clad surface, decreasing the ability to transfer energy. A slight increase in ΔT can put reactor in film boiling region. (0.5)

(2.0) 1.11 When operating in the nucleate boiling region, heat is transferred from the clad to the coolant primarily by bubbles (0.2) formed in nucleation sites (imperfections in the clad) (0.3) then swept into the coolant where they collapse (0.2). If the RCS pressure is increase, less bubbles would be formed (higher saturation temperature) (0.5) and hence transferred to the bulk coolant (0.2). There would be a decrease in the rate of heat transfer at the clad coolant interface (0.2). If the heat transfer rate was held constant, the clad surface temperature would have to increase (0.2) due to less efficient heat transfer at the clad coolant interface. (Boundary layer heat transfer coefficient decreases). (0.2)

(3.0) 1.12 a) Pzr pressure (0.2) ≥ 1684 psia (0.2)
or SIAS
Cont pressure (0.2) ≤ 17.1 psia (0.2)

b) Cont pressure (0.2) ≤ 17.7 psia (0.2)
and an SIAS signal present (0.2) CSAS

c) Cont pressure (0.2) ≤ 17.1 psia (0.2)
Pzr pressure (0.2) ≥ 1684 psia (0.2) CIAS

d) S/G pressure (0.2) ≥ 764 psia (0.2)
Cont pressure (0.2) ≤ 17.1 psia (0.2) MSIS

- END OF SECTION 1 -

2. PROCEDURES - NORMAL, OFF-NORMAL, EMERGENCY, ADMINISTRATIVE AND CONTROLS AND LIMITATIONS

(2.0) 2.1 One of more CEA's (0.1) have not fully inserted into the core (0.1) following a Reactor Trip (0.1).

Shutdown margin (0.1) is determined to be < 5.15% Δ K/K (0.1) when Tavg is > 200°F (0.1).

Shutdown margin (0.1) is determined to be < 2% Δ K/K (0.1) when Tavg is LTE 200°F (0.1).

Unexplained (0.1) positive reactivity addition (0.1).

Uncontrolled (0.1) cooldown of the RCS (0.1) caused by excessive feedwater flow (0.1) or excessive steam withdrawal (0.1).

Keff > .95 (0.15) or boron concentration < 1720 ppm (0.15) with the reactor vessel head closure bolts not fully tensioned (0.1) or with the head removed (0.1).

(0.5) 2.2 Pressurizer level cannot be maintained (0.25) with available charging pumps (0.25).

(0.5) 2.3 c) Increasing containment radition levels.

(1.0) 2.4 A Reactor Trip is required or has occurred. (0.5)

<u>OR</u>

Engineering Safety Features Actuation Signals are present on 2 out of 4 channels. (0.5)

(1.5) 2.5 (0.5 each)

An event which cannot be diagnosed

Multiple Events

Events where the Optimal Recovery Procedure is not maintaining the criteria specified on the Safety Function Status Checklist.

(1.0) 2.6 a) MFW reg valves shut (0.1)
MFW S/U reg valves go to 5% flow position (0.1)
Mn Fd Pumps to go minimum speed (0.1)

b) Automatically reduce FW flowrate (0.1) to remove decay heat (0.1) without overfeeding (0.1).

c) If the controller for that station (0.2) was in manual (0.2).

(1.0) 2.7 Low Steam Generator Level (0.25) of 27.4% (0.1) NR (0.1) coincident (0.1) with no low steam generator pressure (0.25) of GT 764 psia (0.1) on 2/4 channels (0.1).

(1.5) 2.8 a) DNBR (0.25), 1.20 (0.25)

b) Peak Linear Heat Rate (0.25), 21.0 KW/FT (0.25)

c) RCS (0.25), 2750 (0.25)

(2.0) 2.9 (0.5 each)

a) The instantaneous amount of reactivity by which the reactor is subcritical or would be subcritical from its present condition assuming:

- No change in part-length CEA position

AND

- All full length CEA's (shutdown and regulating) are fully inserted except for the single assembly of highest reactivity worth which is assumed to be fully withdrawn.

b) Modes 1-4 shall be GTE 5.15%
Mode 5 shall be GTE 2%

c) In Mode 5 (LT 200°F) there is no cooldown and resultant positive reactivity addition from a postulated Main Steam Line Break as there is from Modes 1-4.

d) By verifying that CEA group withdrawal is within the Transient Insertion Limit Specification (CEA's above Transient Insertion Limits).

(1.0) 2.10 (0.2 each)

a) True
b) False
c) False
d) False
e) True

(1.5) 2.11 Must be approved by two members of the Plant Management (0.25) one of which holds an SRO license on Waterford-3 (0.25).

Can not change the intent of the procedure (0.5).

Must be reviewed by PORC and approved by Plant Manager - Nuclear (0.25) within 14 days of Temporary Approval (0.25).

(2.0) 2.12

Position	Number of Individuals Required Modes 1,2,3 & 4	Modes 5 & 6
SS-SRO License	1	1
SRO (CRS) - SRO License	1	0
RO (NPO) - RO License	2	1
AO	2	1
STA	1	0

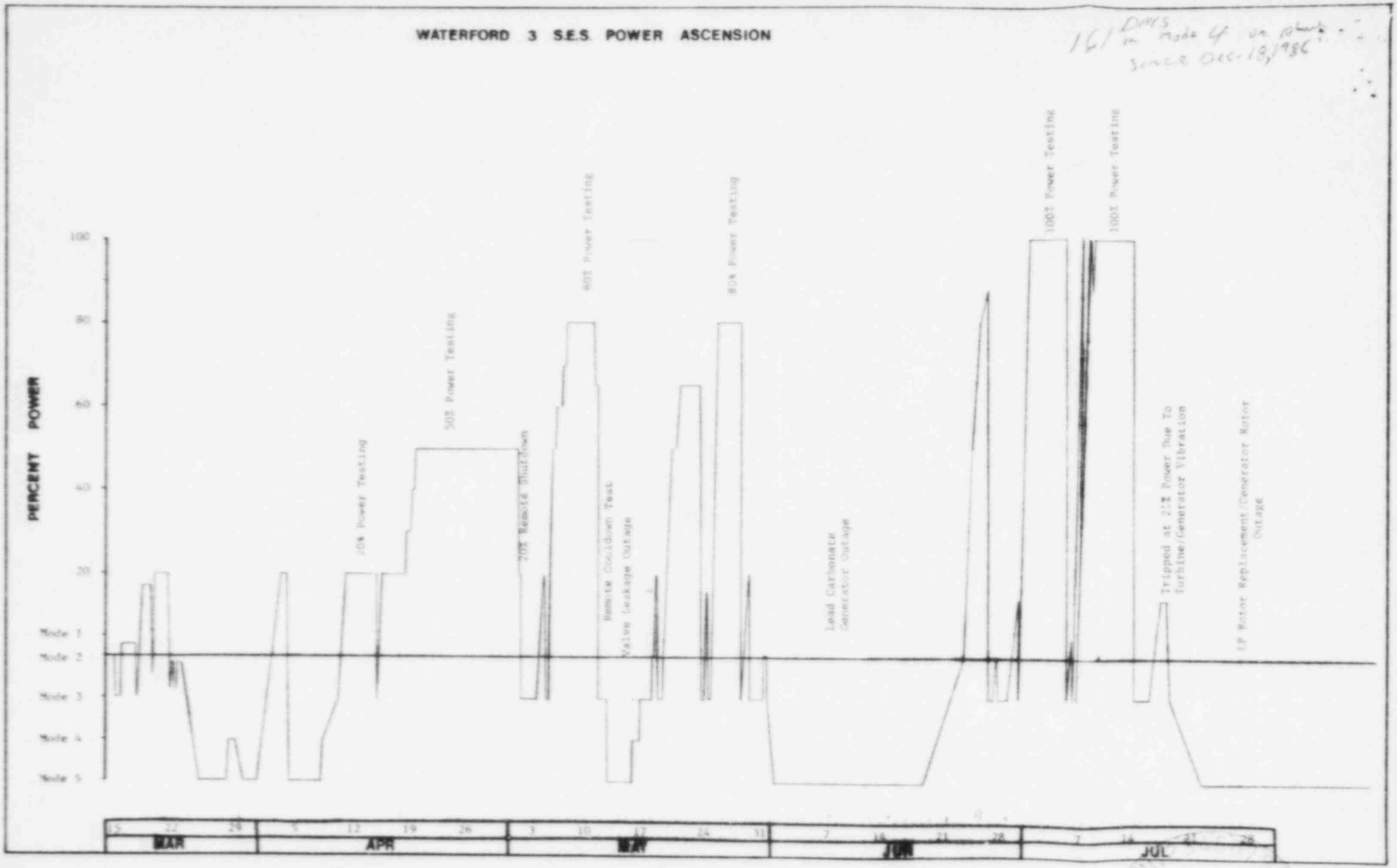
.1 For Position (0.5)
.1 For License (0.3)
.1 For Number of Individuals (0.8)
.2 For Mode (0.4)

(1.0) 2.13 (0.25 each)

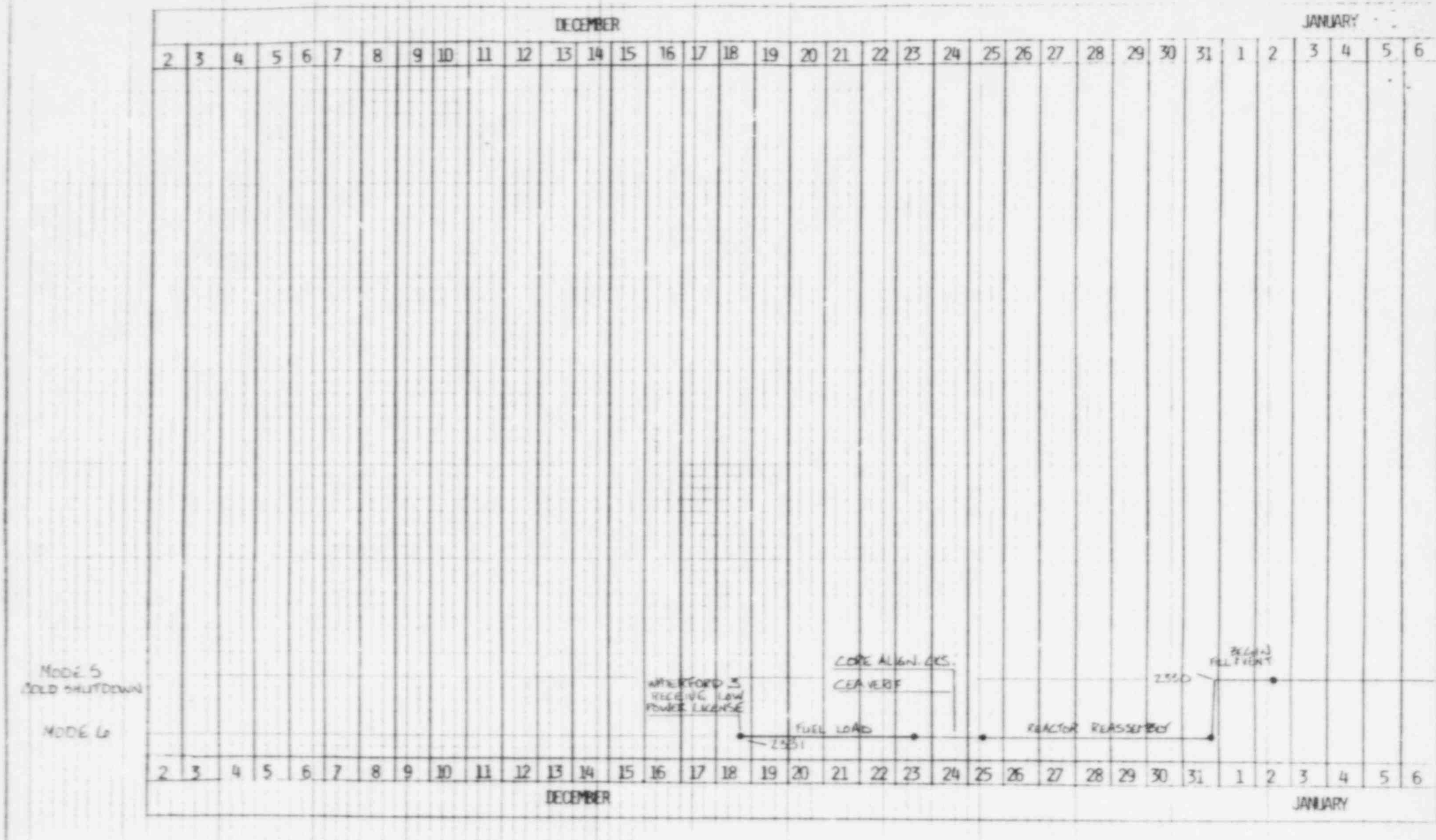
S1 - P4
S2 - P1
S3 - P5
S4 - P2

- END OF SECTION 2 -

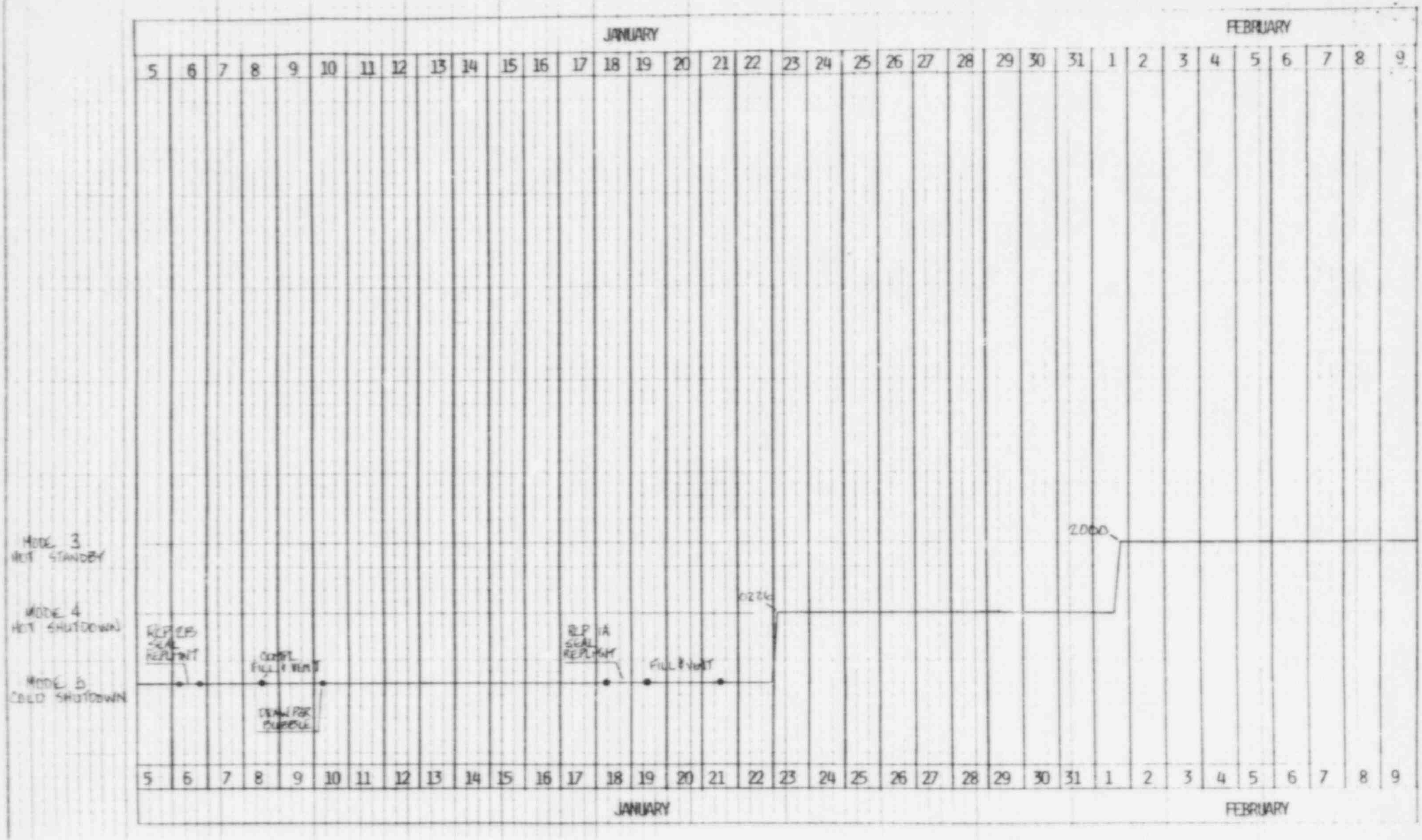
- END OF ANSWER KEY -



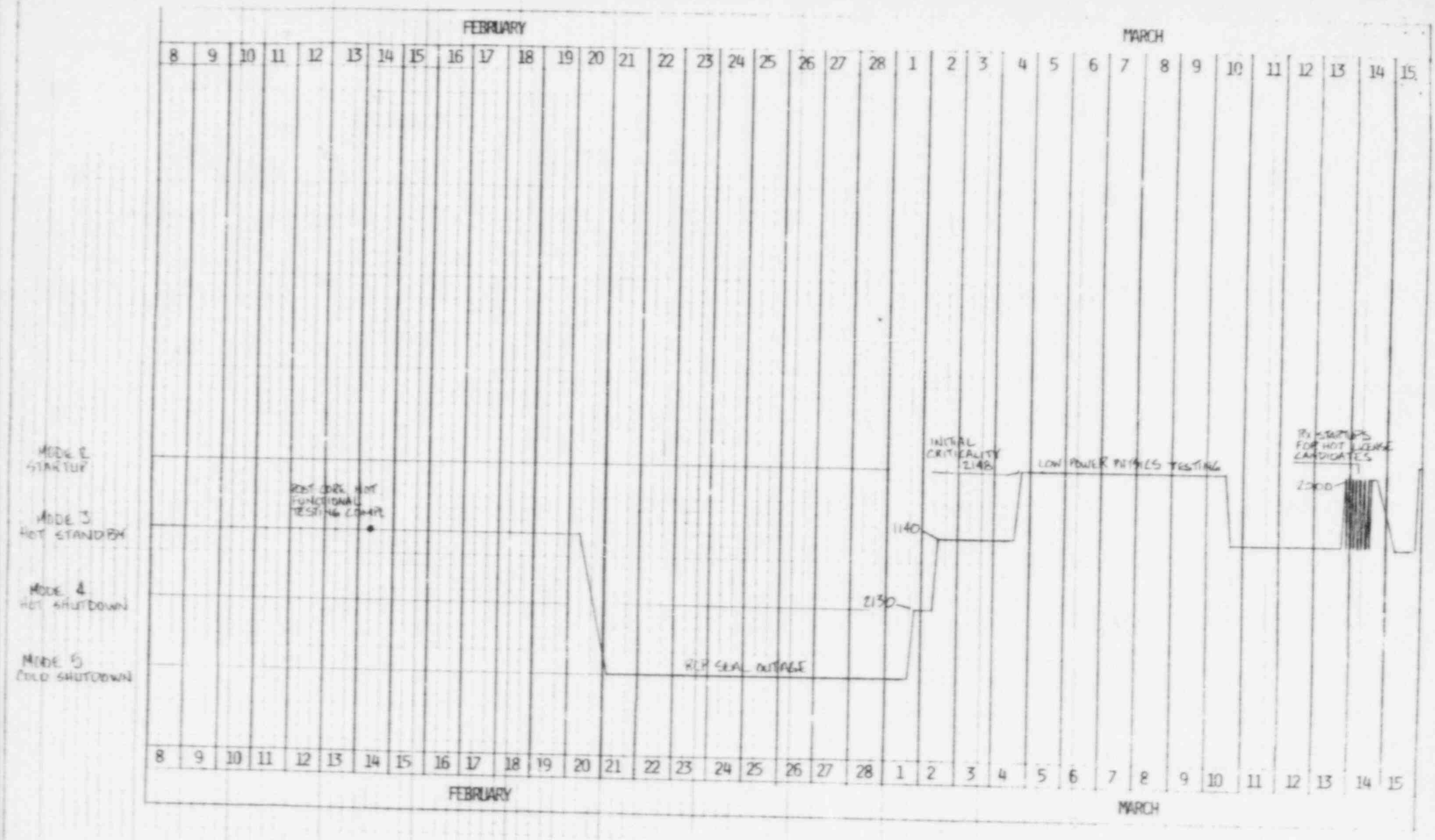
Attachment 4 (Page 1 of 8)



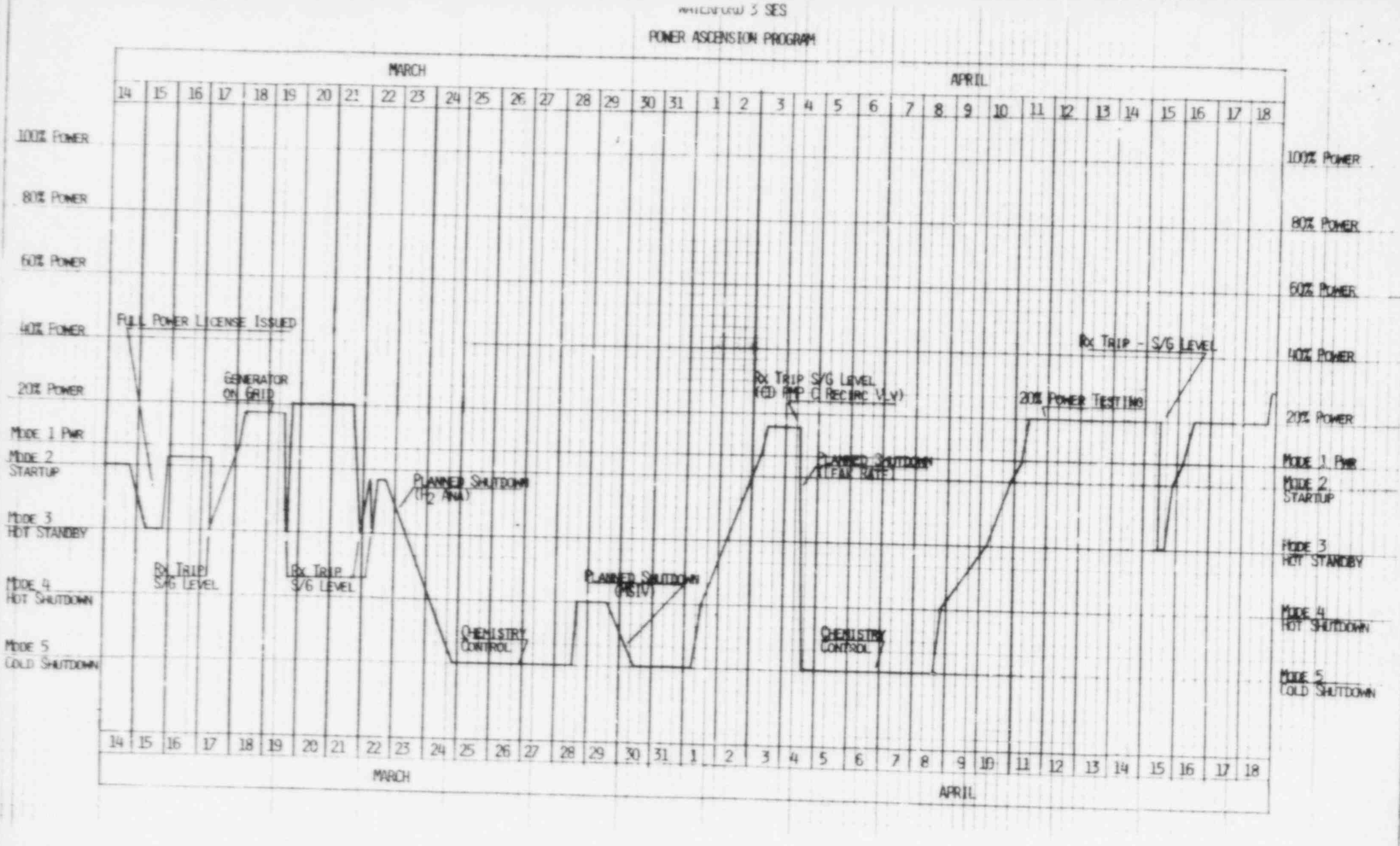
DECEMBER
JANUARY
2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29 30 31 1 2 3 4 5 6
MODE 5
COLD SHUTDOWN
MODE 6
WATERFORD 3
RECEIVE LOW
POWER LICENSE
2331
FUEL LOAD
CORE ALIGN. CKS.
CEA VERIF
REACTOR REASSEMBLY
2330
BEGIN
FILL & VENT



Attachment 4 (Page 3 of 8)

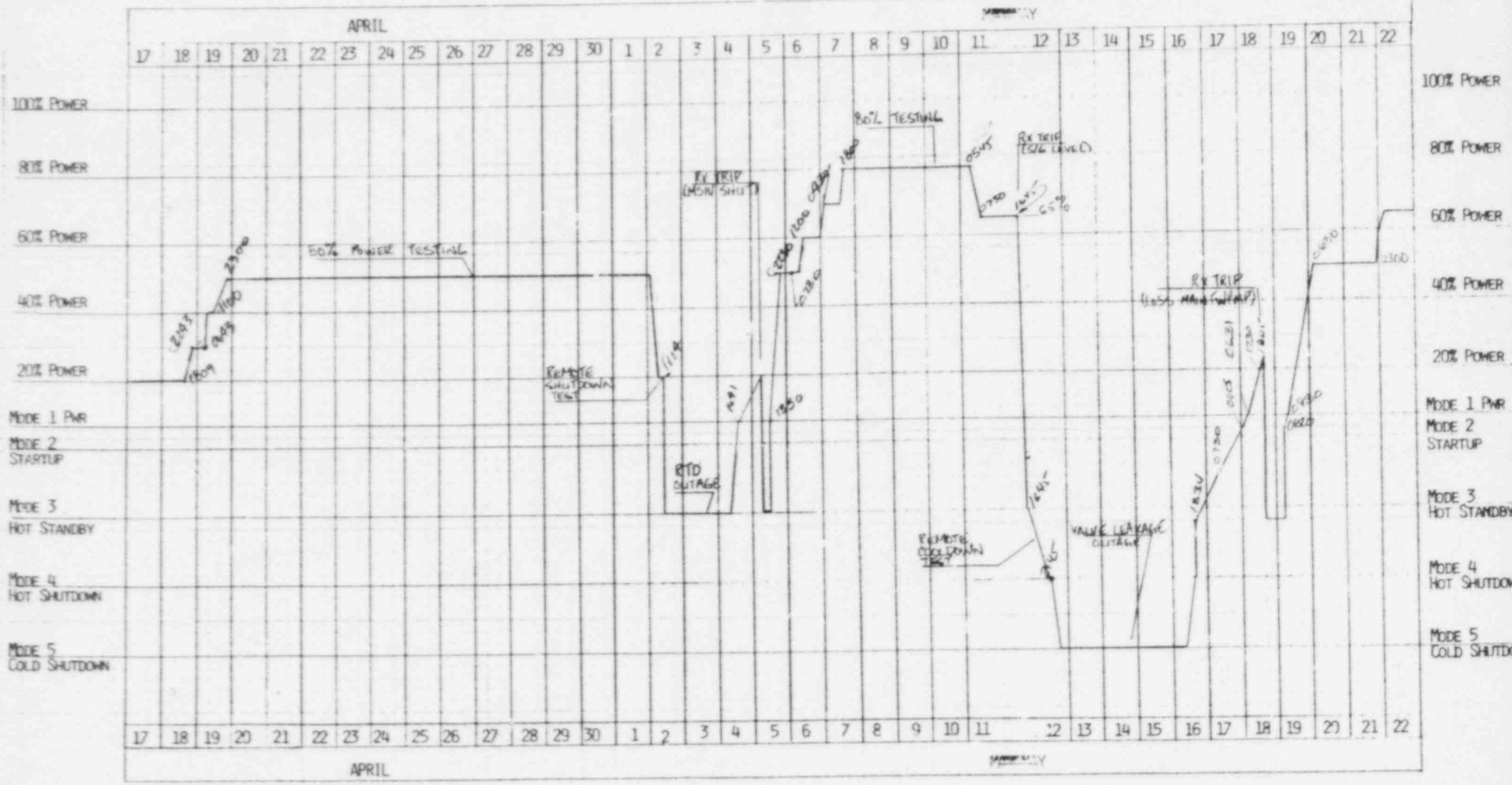


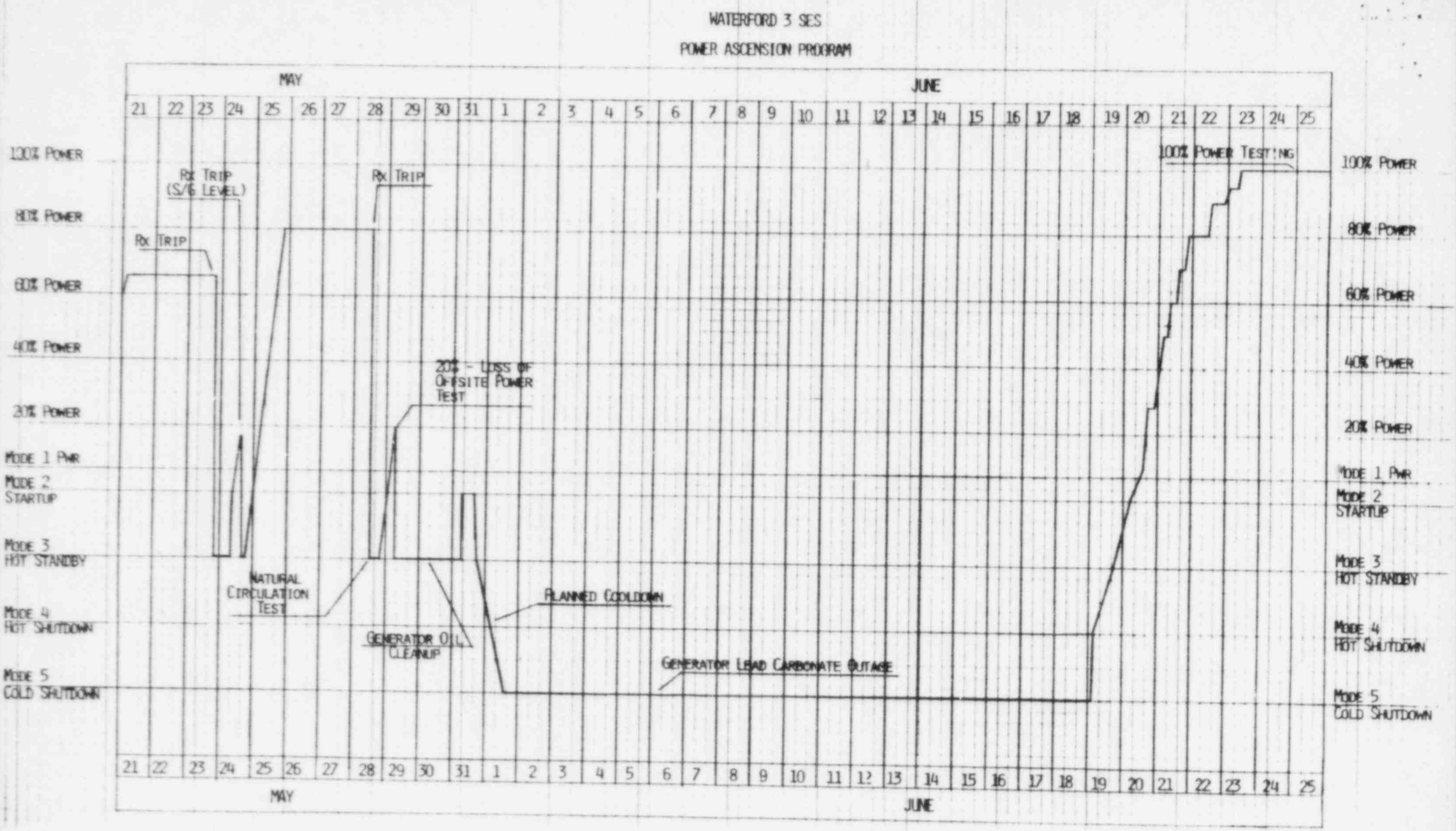
FEBRUARY
MARCH
8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28
1 2 3 4 5 6 7 8 9 10 11 12 13 14 15
MODE 2 STARTUP
MODE 3 HOT STANDBY
MODE 4 HOT SHUTDOWN
MODE 5 COLD SHUTDOWN
POST CORE HOT FUNCTIONAL TESTING COMPL
RCP SEAL OUTAGE
2130
1140
INITIAL CRITICALITY 2148
LOW POWER PHYSICS TESTING
2300


WATERFORD 3 SES
POWER ASCENSION PROGRAM
MARCH
APRIL
100% POWER
80% POWER
60% POWER
40% POWER
20% POWER
MODE 1 PWR
MODE 2 STARTUP
MODE 3 HOT STANDBY
MODE 4 HOT SHUTDOWN
MODE 5 COLD SHUTDOWN
FULL POWER LICENSE ISSUED
GENERATOR ON GRID
RX TRIP S/G LEVEL
RX TRIP S/G LEVEL
PLANNED SHUTDOWN (H_2 ANA)
CHEMISTRY CONTROL
PLANNED SHUTDOWN (MSIV)
RX TRIP S/G LEVEL (FD PMP C RECIRC VLV)
PLANNED SHUTDOWN (LEAK RATE)
CHEMISTRY CONTROL
20% POWER TESTING
RX TRIP - S/G LEVEL

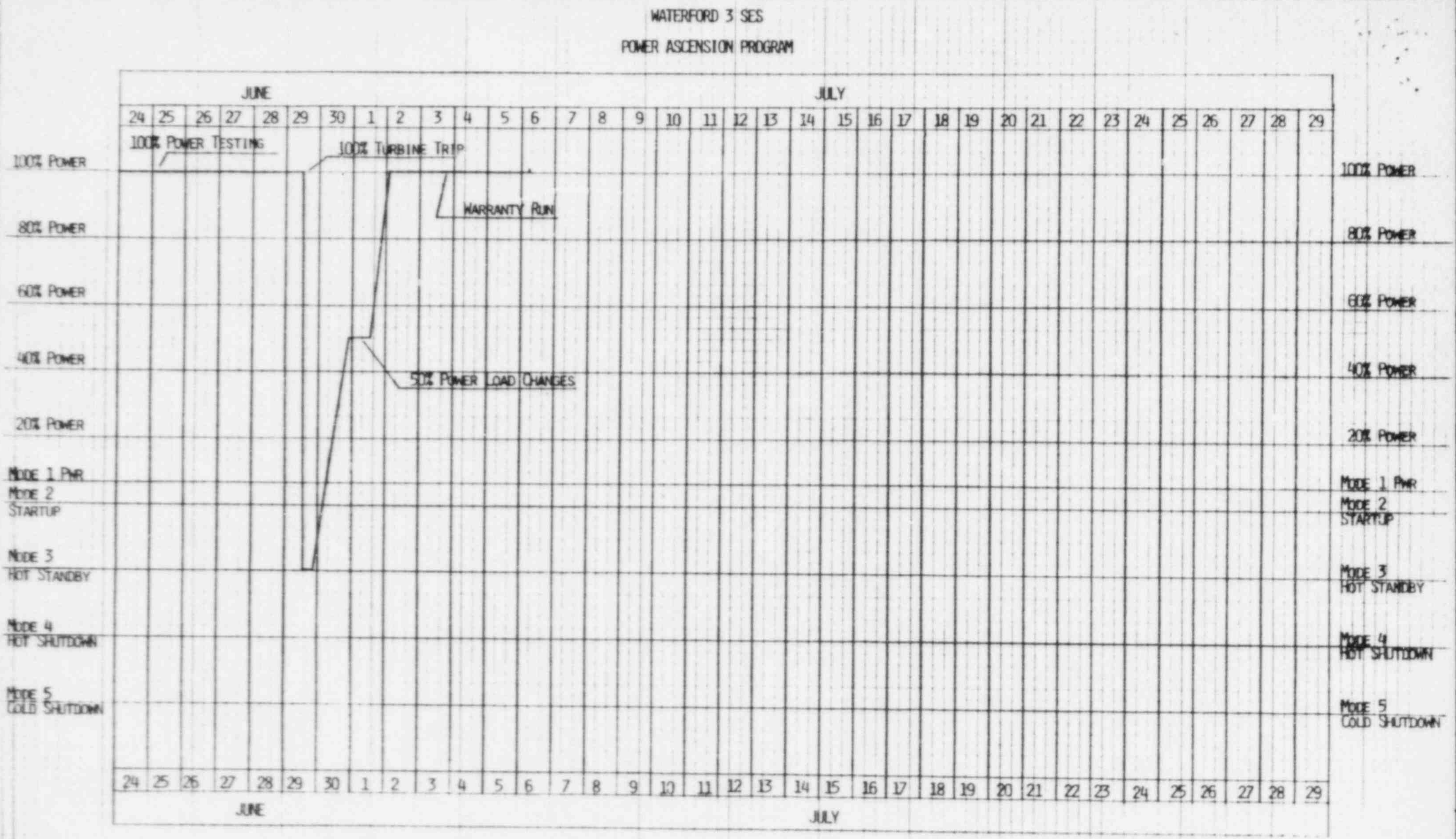
WATERFORD 3 SES

POWER ASCENSION PROGRAM





WATERFORD 3 SES
POWER ASCENSION PROGRAM
MAY
JUNE
21 22 23 24 25 26 27 28 29 30 31 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25
100% POWER
80% POWER
60% POWER
40% POWER
20% POWER
MODE 1 PWR
MODE 2 STARTUP
MODE 3 HOT STANDBY
MODE 4 HOT SHUTDOWN
MODE 5 COLD SHUTDOWN
RX TRIP
RX TRIP (S/G LEVEL)
RX TRIP
NATURAL CIRCULATION TEST
20% - LOSS OF OFFSITE POWER TEST
GENERATOR OIL CLEANUP
PLANNED COOLDOWN
GENERATOR LEAD CARBONATE OUTAGE
100% POWER TESTING


WATERFORD 3 SES
POWER ASCENSION PROGRAM
JUNE
JULY
24 25 26 27 28 29 30 1 2 3 4 5 6 7 8 9 10 11 12 13 14 15 16 17 18 19 20 21 22 23 24 25 26 27 28 29
100% POWER TESTING
100% TURBINE TRIP
WARRANTY RUN
50% POWER LOAD CHANGES
100% POWER
80% POWER
60% POWER
40% POWER
20% POWER
MODE 1 PWR
MODE 2
STARTUP
MODE 3
HOT STANDBY
MODE 4
HOT SHUTDOWN
MODE 5
COLD SHUTDOWN