PDR-016

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

November 4, 1982

Mr. Andrew Steckel
Union of Concerned Scientists
1346 Connecticut Avenue, N.W.
Suite 1105
Washington, DC 20036

IN RESPONSE REFER
TO FOIA-82-503

Dear Mr. Steckel:

This is in response to your letter dated October 21, 1982, in which you requested, pursuant to the Freedom of Information Act, eight specified records.

Upon search of pertinent files maintained by the NRC Office of Nuclear Reactor Regulation, the staff provided the following information regarding the eight requested records.

The records identified as numbers one, two, and seven of your request have already been made available at the NRC Public Document Room (PDR) in the appropriate docket files.

The staff cannot locate the records identified at numbers three and six of your request.

With regard to number four, the staff stated that a meeting, rather than a site visit, was scheduled to be held on February 20, 1979. The staff believes this meeting was rescheduled to March 6, 1979. A copy of the summary of the March 6 meeting as well as the records identified at numbers five and eight of your request, as listed on the enclosed appendix, are being placed in the PDR. These three records will be filed in folder FOIA-82-503 under your name.

Sincerely,

J. M. Felton, Director
Division of Rules and Records
Office of Administration

Enclosure: Appendix

Re: FOIA-82-503

Appendix

1. 4/5/79 Summary of Meeting held on March 6, 1979, w/enclosures. (24 pages)

2. 8/16/79 Letter to Denton from Drake, SCE, "Docket Nos. 50-361 and 50-362, San Onofre Nuclear Generating Station Units 2 and 3", w/attached report, "Evaluation and Action Plan for San Onofre Nuclear Generating Station Units 2 and 3 Relative to the Three Mile Island Incident", August, 1979. (49 pages)

3. 7/13/82 Arizona Public Service Co. press release, "APS Announces Results of Palo Verde Cost and Schedule Analysis". (2 pages)

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

APR 5 1979

DOCKET NO. 50-382

APPLICANT: LOUISIANA POWER AND LIGHT COMPANY

FACILITY: WATERFORD STEAM ELECTRIC STATION, UNIT 3

SUBJECT: MEETING HELD ON MARCH 6, 1979

A meeting was held in Bethesda on March 6, 1979 with the applicant. The purpose of the meeting was to discuss the OL safety review schedule and other related matters. An attendance list is enclosed (Enclosure 1).

Safety Review Schedule

Enclosure 2 is a list of the major milestones of the review process and the tentative dates by which these milestones are to be completed, as presented by the staff at the meeting. The applicant objected to the projected September 1981 issuance of an operating license because it is four months later than the applicant's estimate of completion of construction of the plant. The staff noted that, based on statistical data concerning construction of other plants, Waterford 3 would probably not be ready to load fuel until December 1981.

In support of its contention that the plant would be ready for fuel loading by May 1981, the applicant presented information on plant construction progress. Enclosure 3 is a copy of the slides that were shown.

The applicant noted that, at present, the only significant area in which construction was not on schedule is that of installation of electric cables. Of the four million lineal feet of cable required in total, 422,342 feet were scheduled to be installed by January 31, 1979 but only 172,922 feet had been installed. The applicant noted that, by pulling cable at a rate of 200,000 feet per month, they would be back on schedule by January 1, 1980. While conceding that 200,000 feet per month is an unusually high rate, the applicant noted that rates as high as 300,000 feet per month had been maintained for several months at other plants.

The staff noted that its safety review includes a site visit by the staff to view the arrangement of electric cabling. This visit is usually made when cabling is about 80-90% installed. Although it is preferable that this visit be completed prior to issuance of the safety evaluation report, the status of cable installation determines when the visit can be made. If the visit is not made in time to include the results in the safety evaluation report, it will be carried as an open item to be reported on in a supplement to the safety evaluation report.

M4

GD

~~7904190 272~~ / PDR

The staff noted that operating license reviews for several other plants are scheduled for about the same time frame as the Waterford review and, with our limited resources, accelerating the Waterford review to meet the applicant's May 1981 date would be difficult. In fact, only by slipping the schedule for another plant were we able to move the Waterford completion date up to September 1981 from an originally-estimated completion date of March 1982.

The staff noted that the similarity between Waterford and San Onofre Units 2 and 3 might help speed up those areas of review that had already been completed for San Onofre and that are identical for both plants.

We have found that a major factor in completing our review on time is the resolution of items for which the staff's position is different from that of the applicant. We will try to identify such issues as early in the review as possible. We also pointed out that fully-responsive answers to our requests for information tends to reduce the number of items requiring resolution, thereby shortening the overall schedule.

The staff also noted that an internal effort is under way to improve the quality of safety evaluation inputs in order to reduce the time required for editing. This effort may help reduce the elapsed time between receipt of the applicant's responses to second round questions and positions and issuance of the safety evaluation report. We will also periodically re-evaluate our plant review priorities, giving consideration to plant construction progress. Such re-evaluation might result in an acceleration of the Waterford review.

Other Matters

The applicant stated that sample abstracts of preoperational and startup tests would be sent to us about March 12, 1979, with the request that the staff review them to see if their content is responsive to the staff's acceptance review question 423.2. If they are satisfactory, the other abstracts in the FSAR will be patterned after those three. The staff agreed to reviewsthe samples as soon as possible.

Concerning the applicant's request for an extension of the construction completion dates specified in the construction permit, the applicant noted that all the delay factors and delay times affecting the construction critical path are given in the applicant's letter of September 5, 1978.

The staff reiterated what had been told to the applicant earlier that the applicant's earlier request concerning conformance with applieable codes pursuant to 10 CFR 50.55 would be considered in its review of the FSAR rather than be handled separately.

The applicant provided a list (Enclosure 4) of responses to staff questions for which the applicant would like staff feedback as early as possible. These involve matters that, if the applicant's responses are not acceptable, should be resolved soon because of their potential effect on design and construction. The staff stated that it would try to get early resolution.

The applicant stated that the revised fire protection plan would be submitted in May 1979.

The applicant stated that he had been sending copies of all applicant-generated documents to the local public document room (LPDR) and that NRC has been doing the same. Such duplication is not necessary. Later, the staff told the applicant that he should merely monitor the LPDR to help assure that the FSAR and Environmental Report are kept up-to-date.

It was agreed that all technical contact with the applicant would be through Roy Prados. For matters concerning the environmental review, the applicant should contact the NRC environmental project manager, Phillip Cota. All matters concerning the safety review and general licensing concerns should be directed to the NRC licensing project manager, Robert Benedict.

Original signed by
R. A. Benedict

R. A. Benedict
Light Water Reactors
Branch No. 2
Division of Project Management

Enclosures:
1. Attendance List
2. Tentative Schedule of OL Safety Review
3. Copies of 12 Slides
4. List of Responses for Early Feedback

ccs w/enclosures:
See next page

Mr. D. L. Aswell
Vice President, Power Production
Louisiana Power & Light Company
142 Delaronde Street
New Orleans, Louisiana 70174

cc: W. Malcolm Stevenson, Esq.
Monroe & Lemann
1424 Whitney Building
New Orleans, Louisiana 70130

Mr. E. Blake
Shaw, Pittman, Potts and Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

Mr. D. B. Lester
Production Engineer
Louisiana Power & Light Company
142 Delaronde Street
New Orleans, Louisiana 70174

Lyman L. Jones, Jr., Esq.
Gillespie & Jones
910 Security Homestead Building
4900 Veterans Memorial Boulevard
Metairie, Louisiana 70002

Luke Fontana, Esq.
Gillespie & Jones
824 Esplanade Avenue
New Orleans, Louisiana 70116

Stephen M. Irving, Esq.
One American Place, Suite 1601
Baton Route, Louisiana 70825

APR 5 1979

ENCLOSURE 1

ATTENDANCE LIST

WATERFORD UNIT 3
MARCH 6, 1979

LOUISIANA POWER AND LIGHT COMPANY

S. A. Alleman
D. B. Lester
R. W. Prados

EBASCO SERVICES

J. Costello
J. Crnich

COMBUSTION ENGINEERING

C. B. Brinkman
H. B. Mulliken

MONROE AND LEMANN

W. M. Stevenson

SHAW, PITTMAN, POTTS AND TROWBRIDGE

E. L. Blake
A. R. Yuspeh

NRC - STAFF

R. Baer
R. Benedict
W. H. Lovelace
H. J. McGurren
D. Ross
W. J. Ross
R. C. Stewart
D. B. Vassallo

ENCLOSURE 2 APR 5 1979

WATERFORD 3

TENTATIVE SCHEDULE
OL SAFETY REVIEW

Q1's TO APPLICANT

PSB, ICB	6/15/79
ASB	6/25/79
MEB, MTEB, SEB	5/4/79
AB (Reactor Analysis)	5/18/79
A3 (Systems Analysis)	4/13/79
All others	4/13/79

Q1 RESPONSES RECEIVED

PSB, ICB	9/7/79
ASB	8/31/79
MEB, MTEB, SEB	8/10/79
AB (Reactor Analysis)	8/24/79
AB (Systems Analysis)	7/20/79
All others	7/20/79

Q2's/POSITIONS TO APPLICANT	11/9/79
Q2/POSITION RESPONSES RECEIVED	2/15/80

SAFETY EVALUATION ISSUED	11/28/80
ACRS MEETING	1/9/81
SUPPLEMENT TO SAFETY EVALUATION ISSUED	4/10/81
SAFETY HEARING BEGINS	4/28/81
SAFETY HEARING ENDS	6/26/81
ASLB INITIAL DECISION, OL ISSUED	9/18/81

1 —>

ENCLOSURE 3

APR 5 1979

WATERFORD 3 SES 1/31/79 PROGRESS STATUS

CRAFT DATA

1.	TOTAL CRAFT AS OF 1/31/79	2,277
2.	PERCENT COMPLETE AS OF 1/31/79	57.7%
3.	CRAFT MAN-HOURS ESTIMATED THROUGH 1/31/79	6,681,000
	CRAFT MAN-HOURS SPENT THROUGH 1/31/79	6,585,000

BULK QUANTITIES

ITEM	U.M.	FORECAST	SCHEDULED* THROUGH 1/31/79	ACTUAL THROUGH 1/31/79	DATE OF EXPECTED RECOVERY
CONCRETE	CY	205,262	195,797	183,600	NOT REQUIRED
LARGE PIPE	LF	109,999	57,188	59,896	-
SMALL PIPE	LF	126,393	24,477	25,342	-
LARGE HANGERS	EA	6,242	4,519	2,830	11/1/79
CABLE TRAY	LF	43,762	39,717	38,250	PUNCH LISTING
EXPOSED CONDUIT	LF	363,870	84,373	103,610	-
CABLE	LF	4,000,000	422,342	172,922	1/1/80
TERMINATIONS	EA	131,000	-0-	-0-	BEGIN IN MARCH

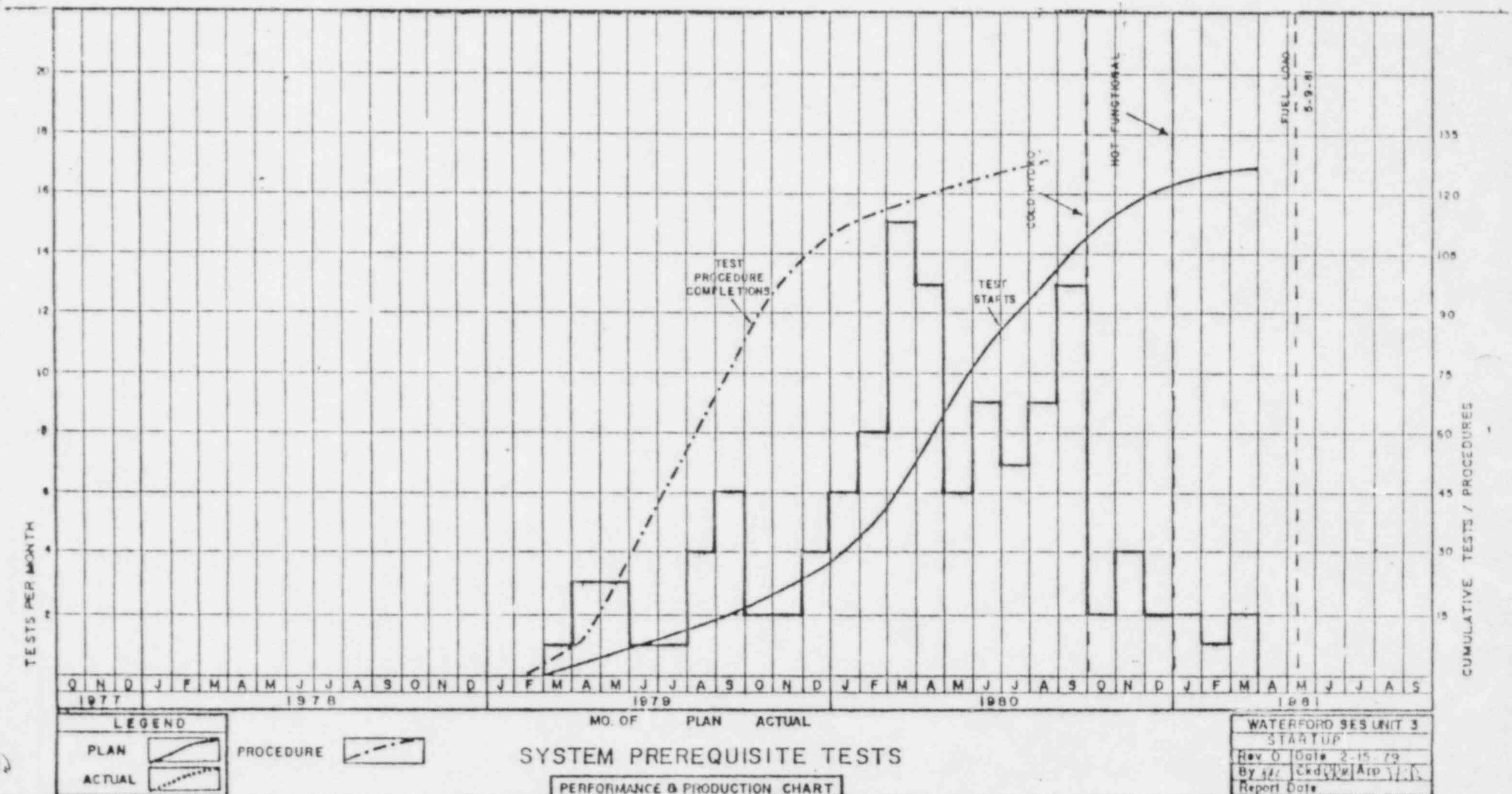
*PER PROJECT SCHEDULE ESTABLISHED 4/78

APR 5 1979

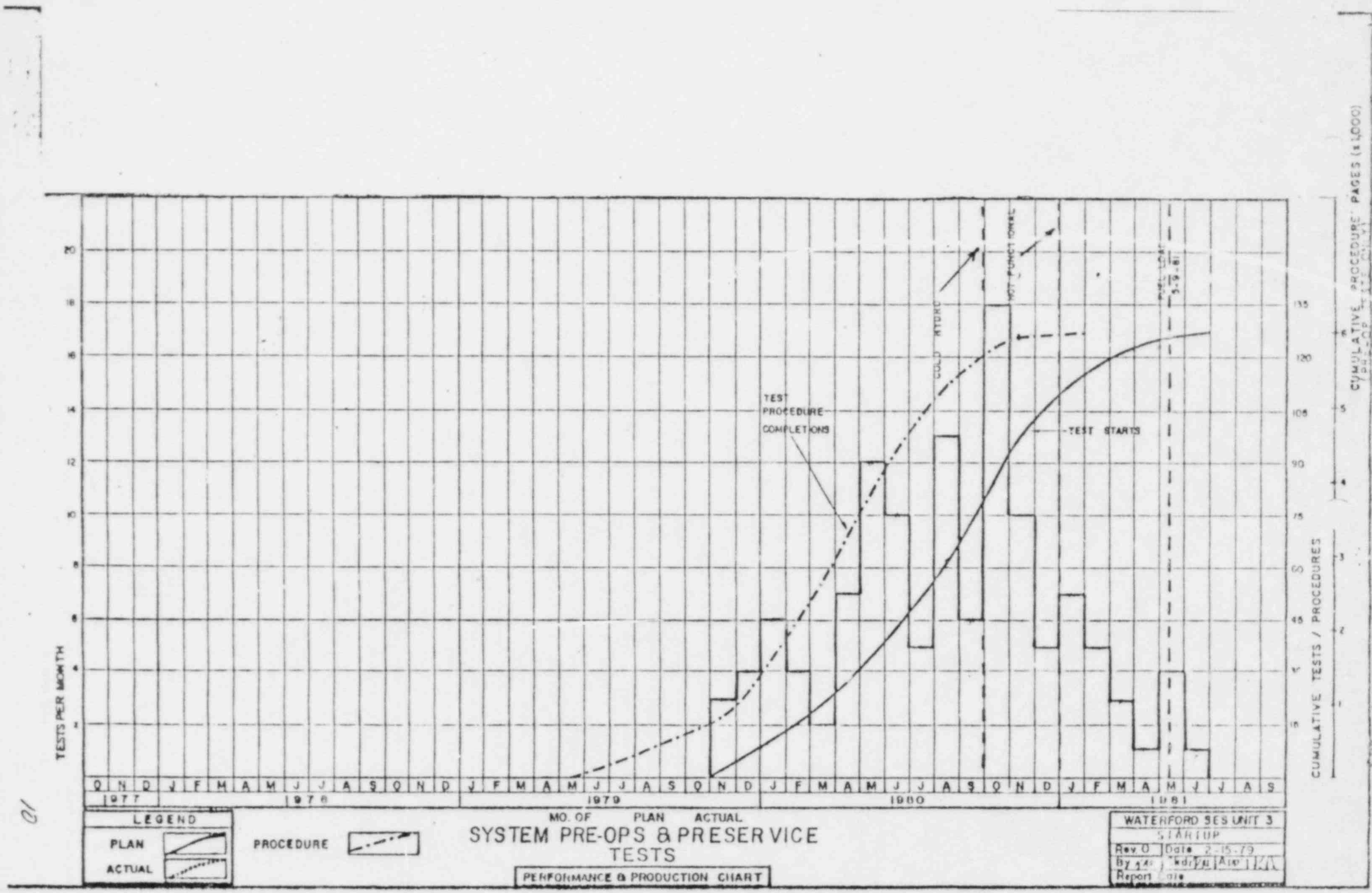
WATERFORD SES UNIT NO 3
DRAWING CLOSE-OUT SCHEDULE PRODUCTION SUMMARY
STATUS OF CLOSED DRAWINGS

1/31/79 STATUS

	January 1979	Cumulative Through January 1979	Scheduled After February 1, 1979
Overall			
Scheduled	1	1469	6
Actual	3	1460	
Reopened	0		
Variance		-9	
Mechanical			
Scheduled	0	233	0
Actual	0	233	
Reopened	0		
Variance		0	
Concrete-Hydraulic			
Scheduled	0	304	0
Actual	0	301	
Reopened	0		
Variance		-3	
Architectural-Structural			
Scheduled	0	517	2
Actual	3	516	
Reopened	0		
Variance		-1	
HVAC			
Scheduled	0	120	0
Actual	0	120	
Reopened	0		
Variance		0	
Electrical			
Scheduled	1	221	2
Actual	0	218	
Reopened	0		
Variance		-3	
I&C			
Scheduled	0	46	2
Actual	0	44	
Reopened	0		
Variance		-2	
Plumbing			
Scheduled	0	28	0
Actual	0	28	
Reopened	0		
Variance		0	

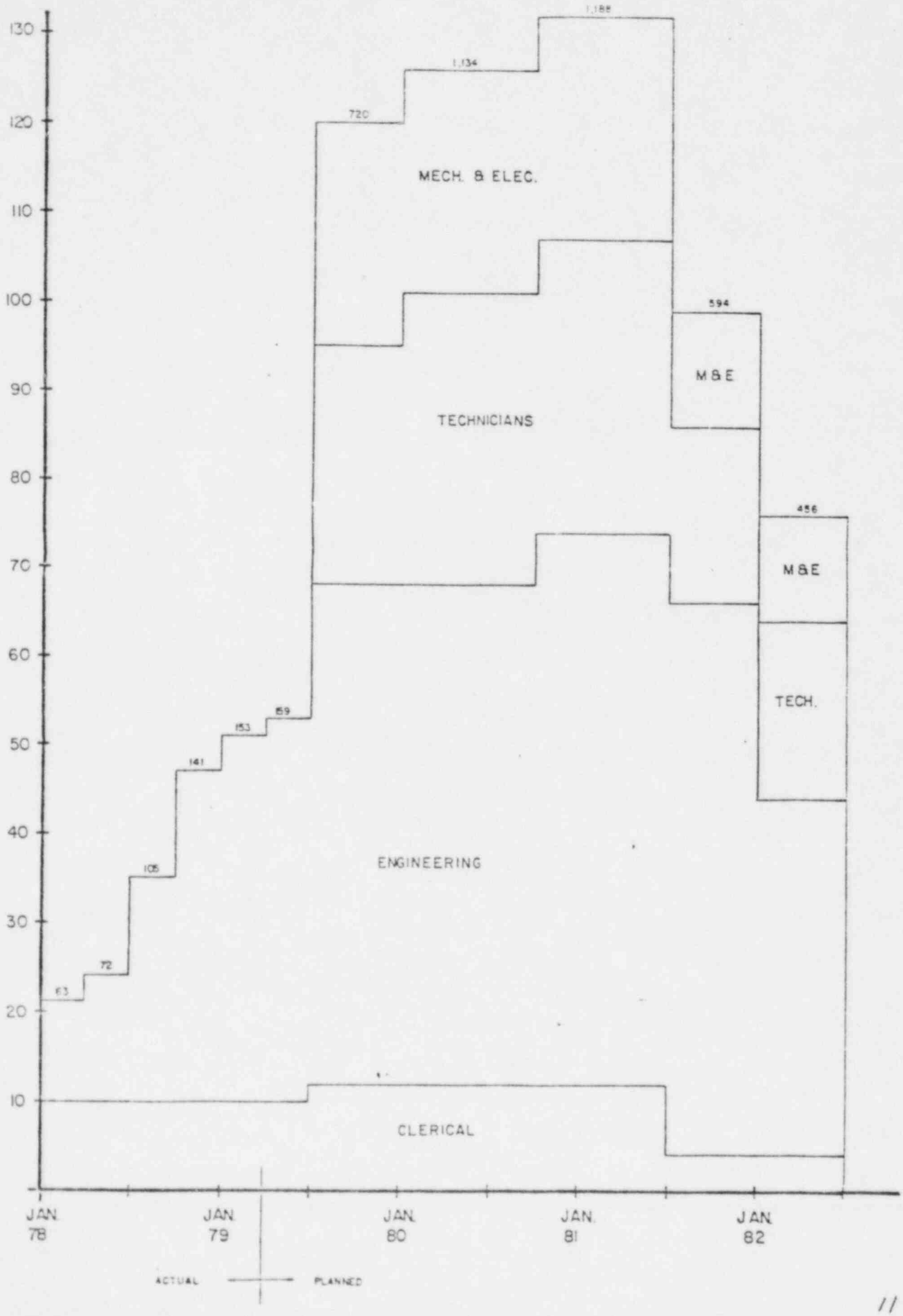


TESTS PER MONTH
CUMULATIVE TESTS / PROCEDURES
TEST PROCEDURE COMPLETIONS
TEST STARTS
COLD HYDRO
HOT FUNCTIONAL
FUEL LOAD 5-9-81
1977
1978
1979
1980
1981
LEGEND
PLAN
ACTUAL
PROCEDURE
MO. OF PLAN ACTUAL
SYSTEM PREREQUISITE TESTS
PERFORMANCE & PRODUCTION CHART
WATERFORD SES UNIT 3
STARTUP
Rev 0 Date 2-15-79
Report Date


TESTS PER MONTH
CUMULATIVE TESTS / PROCEDURES
CUMULATIVE PROCEDURE PAGES (x1,000)
TEST PROCEDURE COMPLETIONS
COLD HYDRO
HOT FUNCTIONAL
TEST STARTS
FUEL LOAD 5-9-81
1977
1978
1979
1980
1981
LEGEND
PLAN
ACTUAL
PROCEDURE
MO. OF PLAN ACTUAL
SYSTEM PRE-OPS & PRESERVICE TESTS
PERFORMANCE & PRODUCTION CHART
WATERFORD SES UNIT 3
STARTUP
Rev. 0
Date 2-15-79
Report Date

TOTAL: 4,785 MM = 105,270 MD

APR 5 1979



11

STATUS OF MAJOR CONTRACTORS

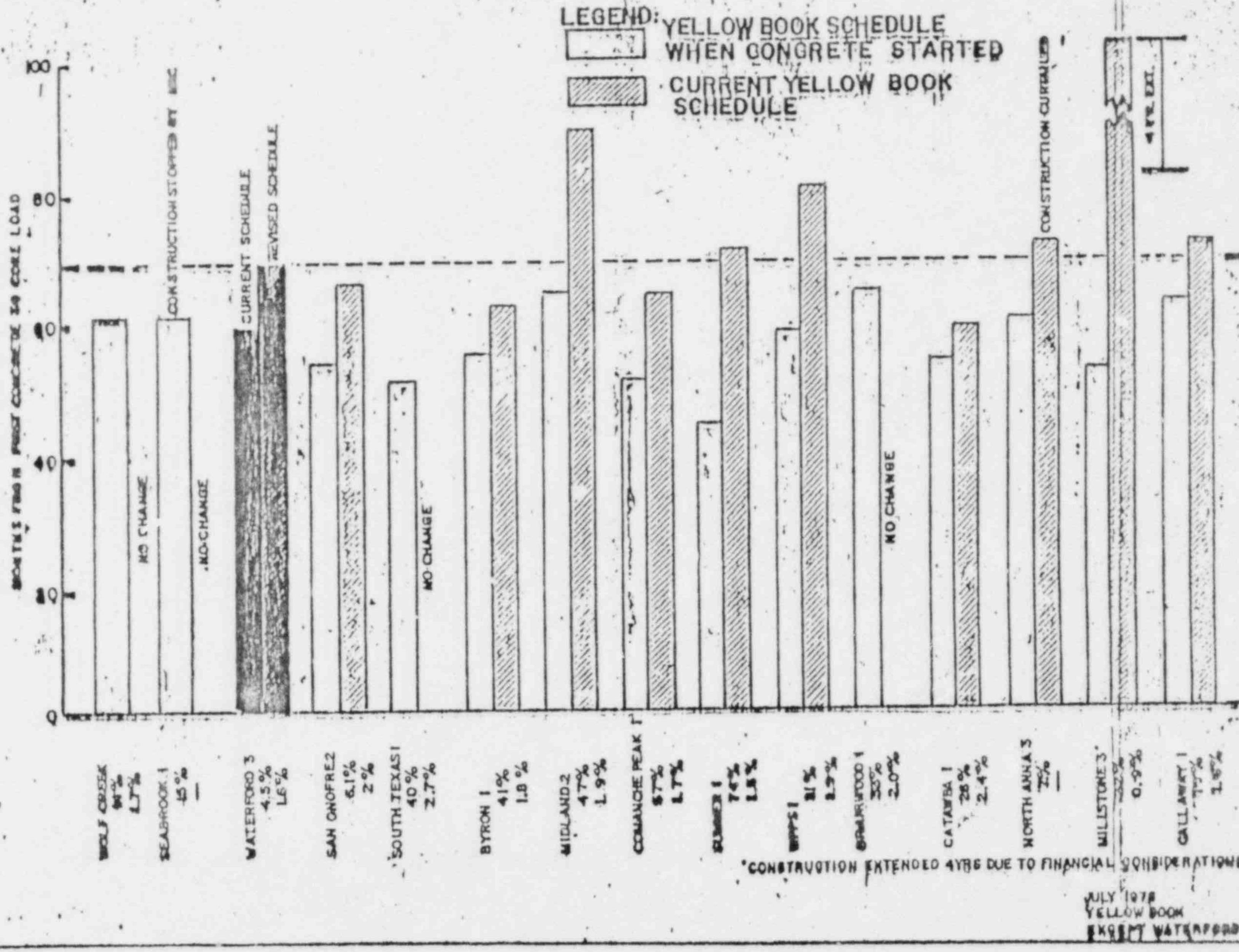
1/31/79

CONTRACTOR	RESPONSIBILITY	9/1/78 % COMPLETE	1/31/79 % COMPLETE
TOMPKINS-BECKWITH	PROCESS PIPING	21.7*	36
FISCHBACH & MOORE	ELECTRICAL	25	39
WALDINGER	HVAC	25	48
NISCO	NSSS INSTALLATION	0	16
J. A. JONES	NUCLEAR ISLAND CONCRETE	93	97
TELLEPSEN	WATERFRONT FACILITIES	30	68

*CORRECTED FROM 23.0%

SCHEDULE COMPARISON
PWR'S SCHEDULED FOR FUEL LOAD 1979-1982
YELLOW BOOK CONSTRUCTION DURATION: FIRST CONCRETE TO FUEL LOAD

APR 5 1979

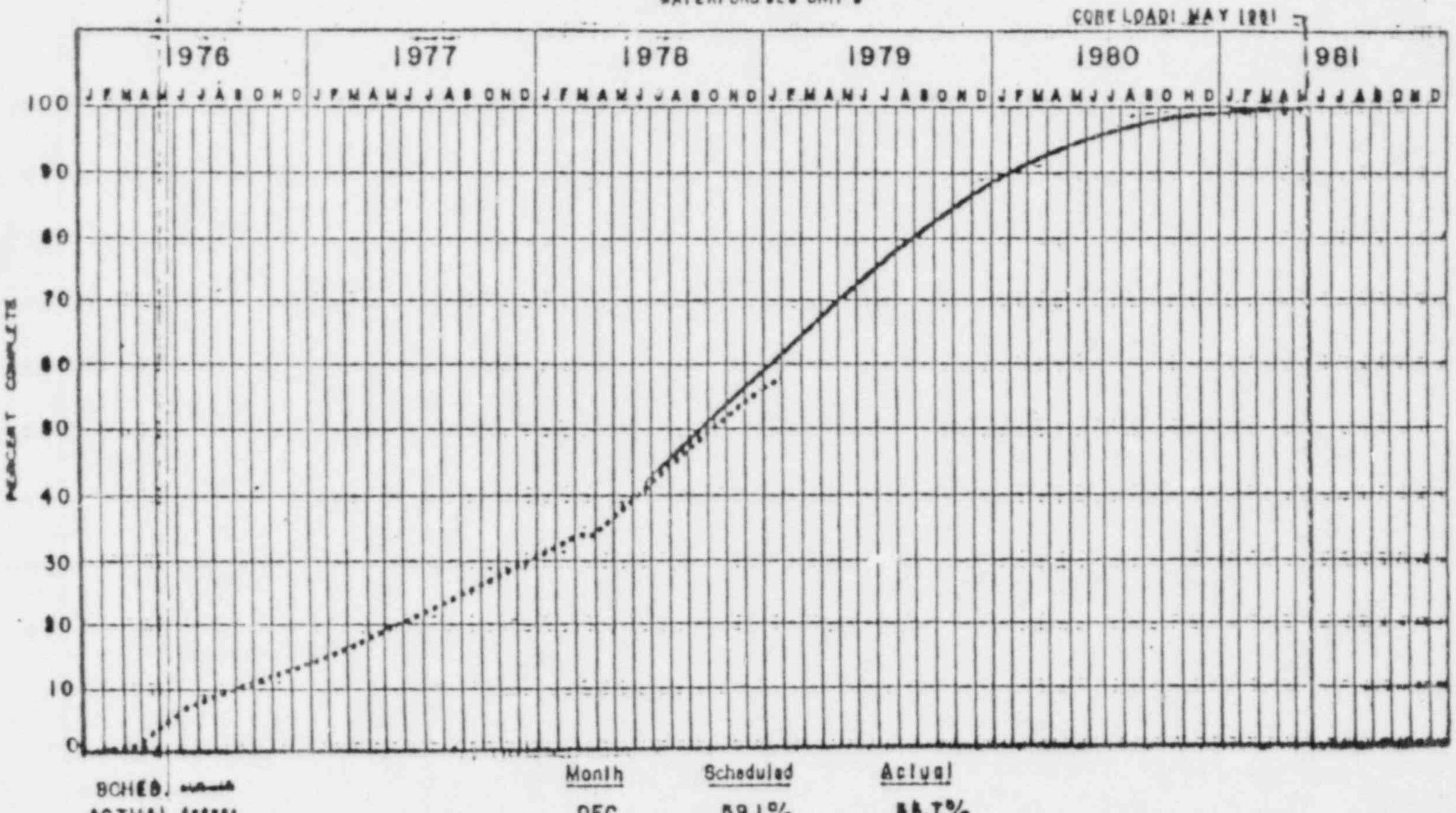


W-3 Schedule compares favorably with other comparable units.

— — fuel loading.

CONSTRUCTION ACCOMPLISHMENTS

LOUISIANA POWER & LIGHT COMPANY
WATERFORD SES UNIT 3



Month	Scheduled	Actual
DEC	59.1%	55.7%
JAN	61.9%	57.7%

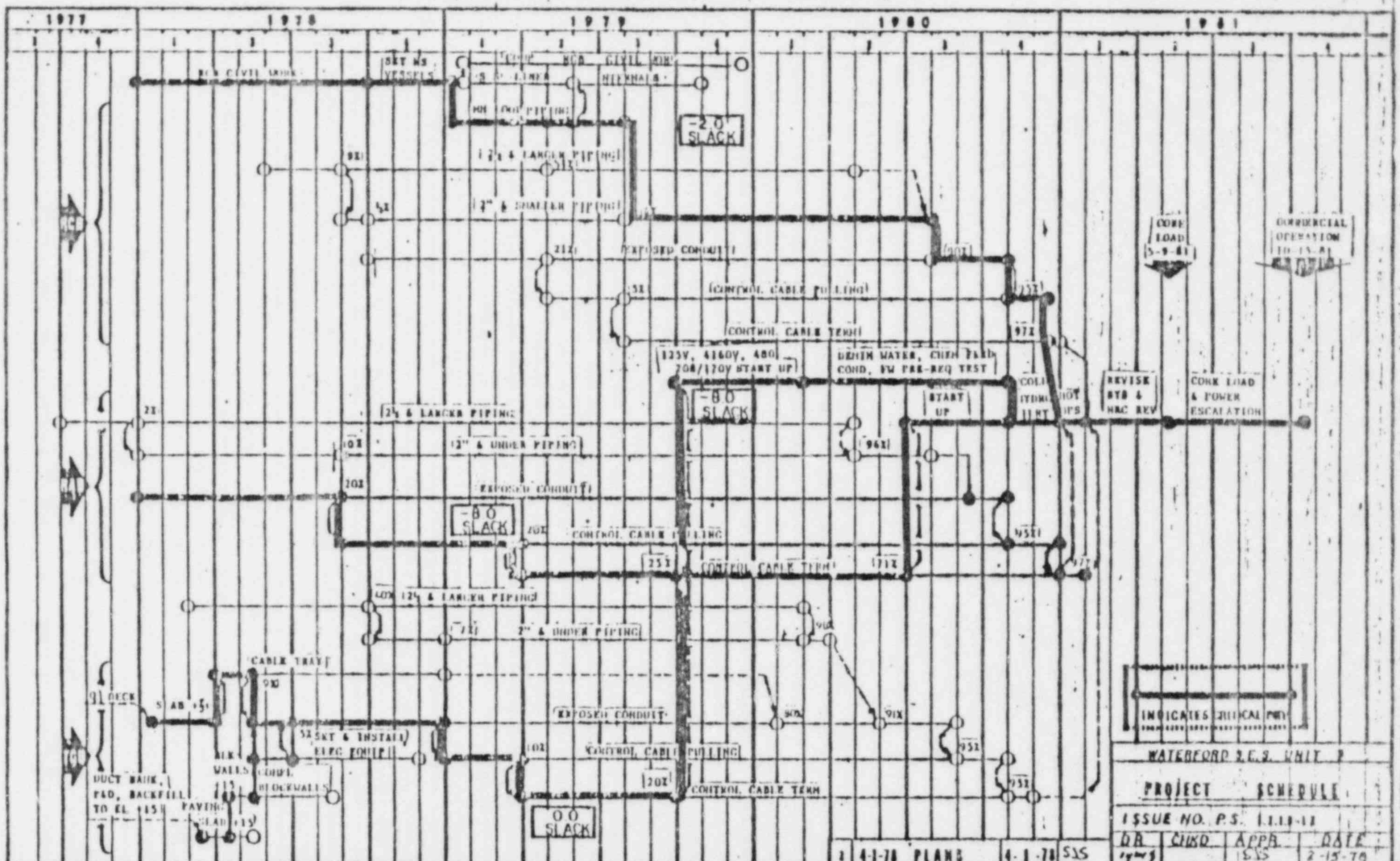
APR 5 1979

LAST 3 MONTH YELLOW BOOK REPORTED CONSTRUCTION ACCOMPLISHMENTS

	NOV. YELLOW BOOK	FEB. YELLOW BOOK	% ACCOMPLISHMENTS
WOLF CREEK	24	27	3%
SAN ONOFRE	70	74	4%
BRAIDWOOD	42	45	3%
CATAWBA	26	40	14%
WATERFORD	51	56	5%

APR 5 1979

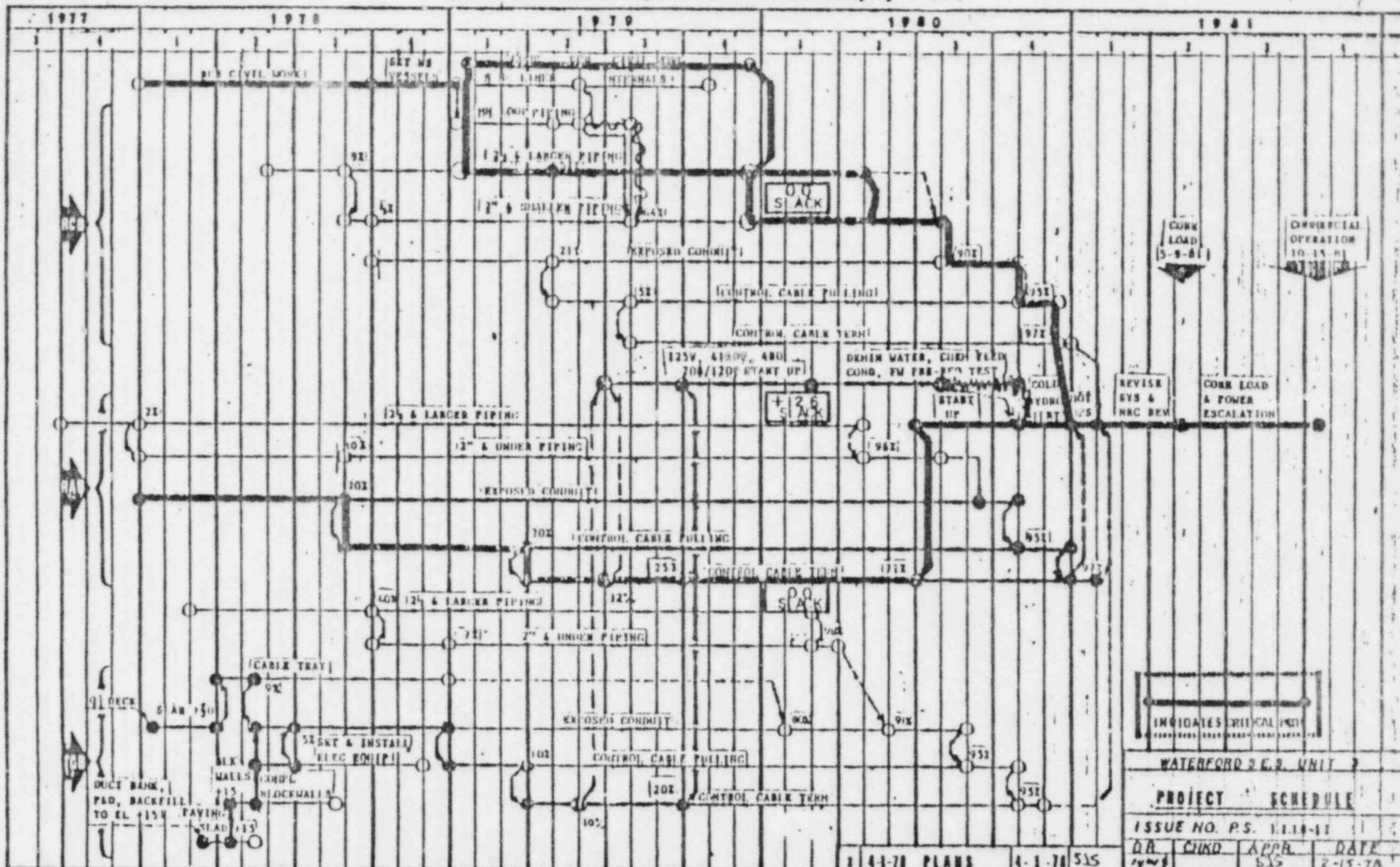
FEB 1, 1979: PROJECT STATUS (Based on 4/1/78 logic)



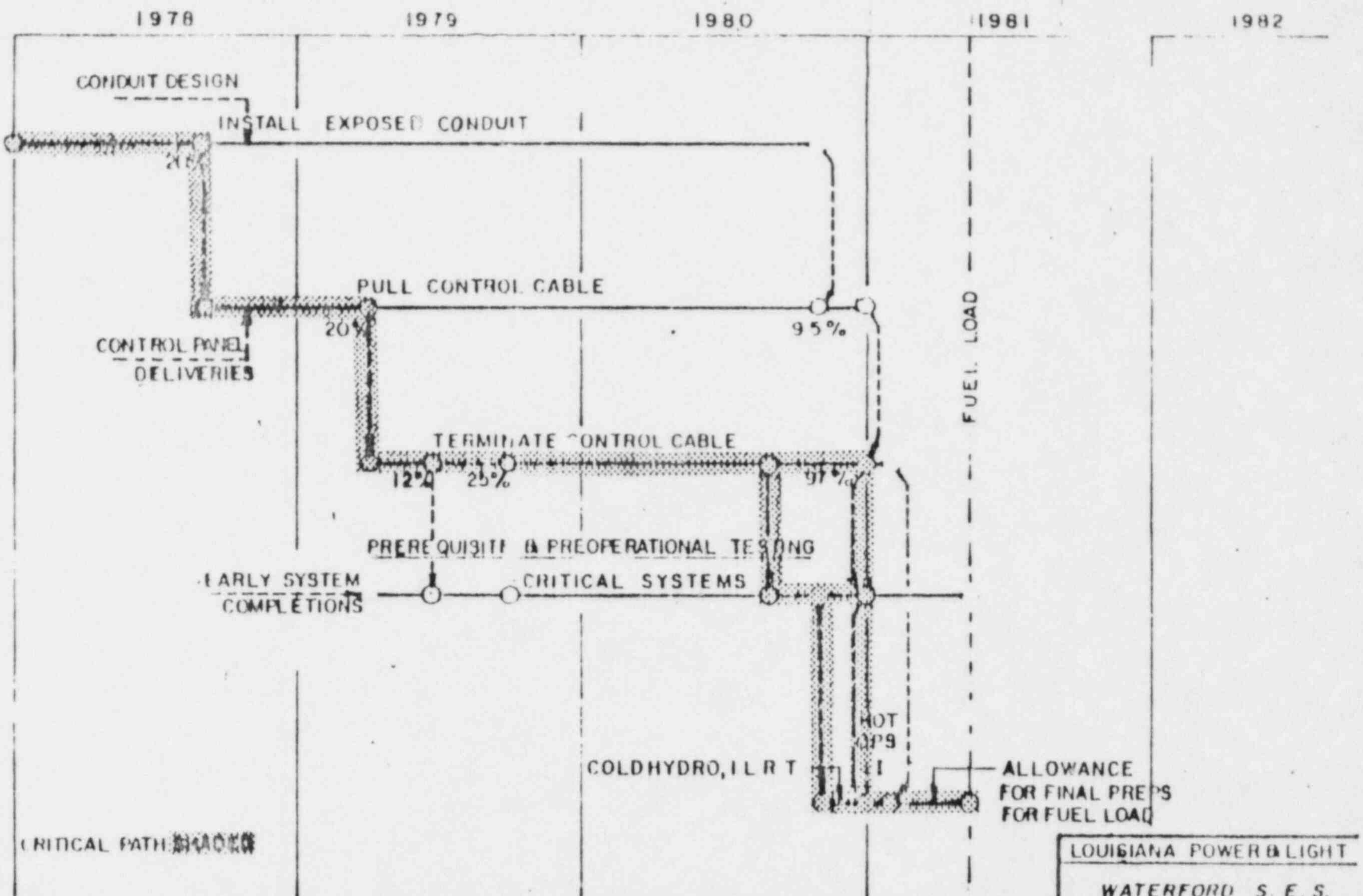
16

PROJECTED 1/1/80 PROJECT STATUS

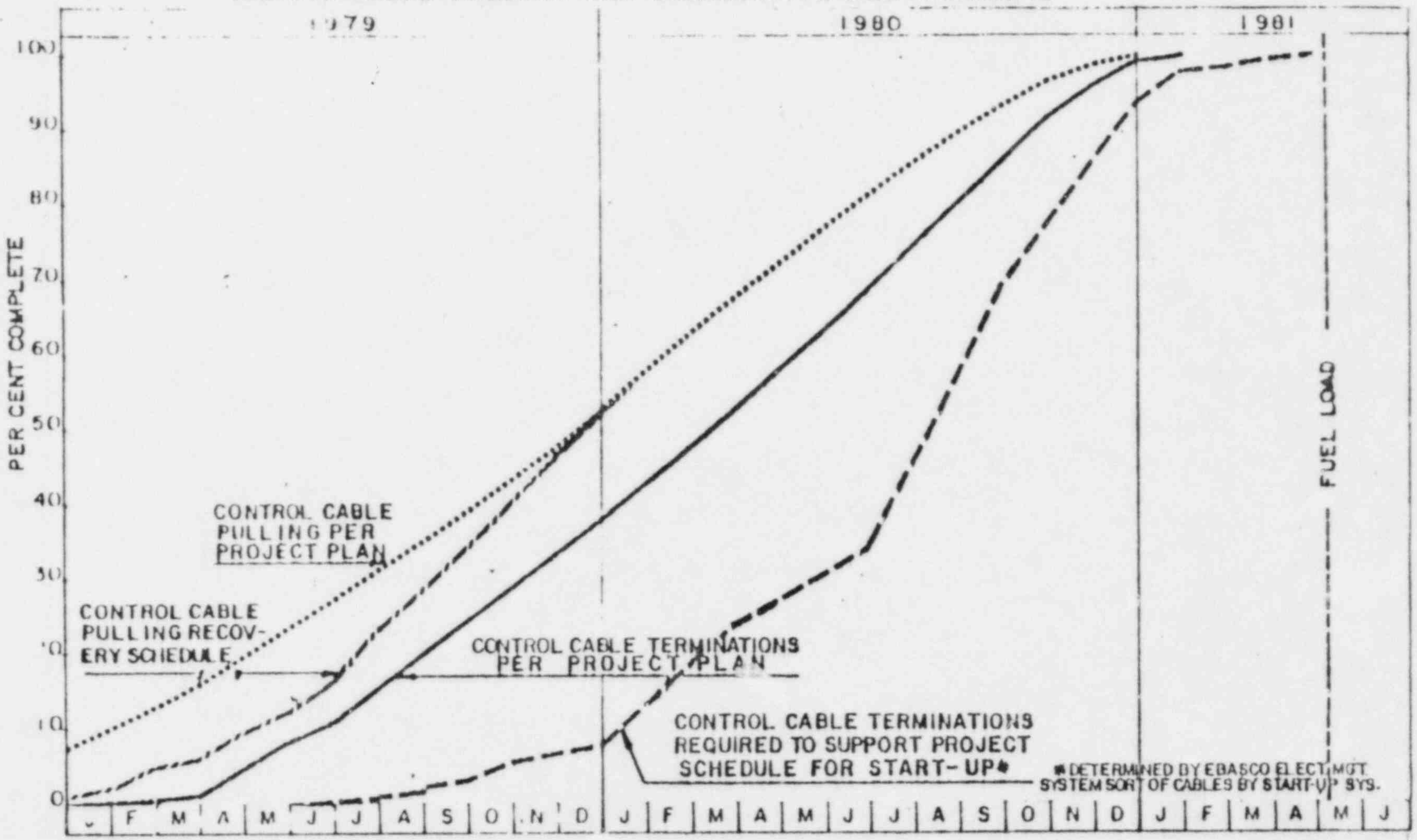
(Based on current logic & Predicted Recovery by 1/80)



ELECTRICAL RESTRAINTS ON START-UP



CONTROL CABLE PULLING AND TERMINATION REQUIREMENTS



APR 5 1979

19

APR 5 1979

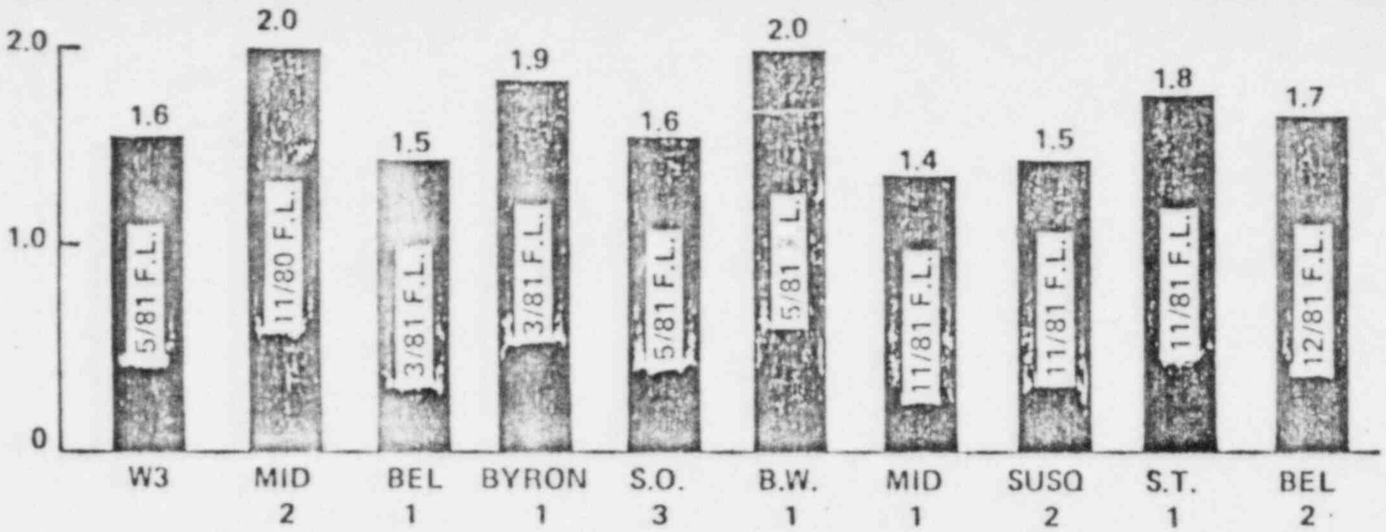
JANUARY PERCENT COMPLETE VARIANCE

SCHEDULE CRITICAL CONTRACTS

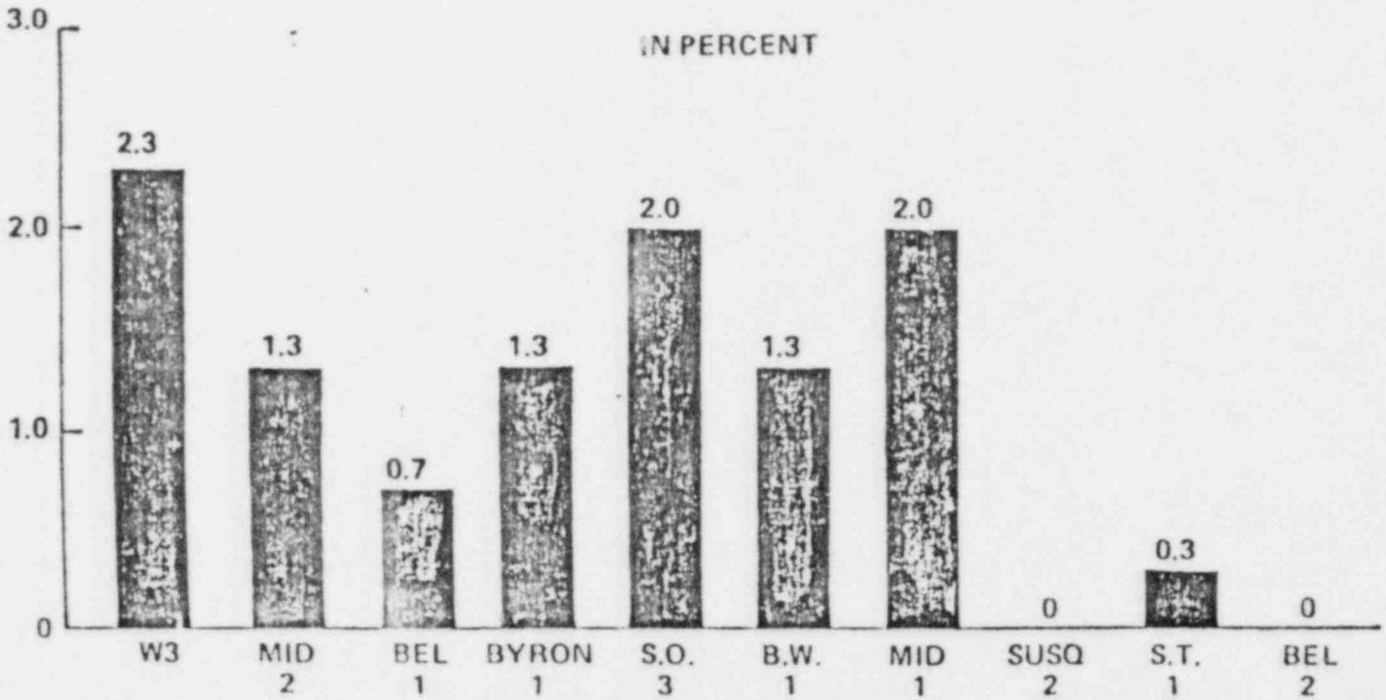
	DEC.	JAN.	△	WT. FACT.	% COMPLETE
FISCHBACH & MOORE	(1.13)	(1.27)	(0.14)	17.5	39.0
WALDINGER	(0.18)	(0.20)	(0.02)	7.2	47.7
NISCO	(0.12)	(0.09)	0.03	1.9	16.3
MERCURY	(0.08)	(0.12)	(0.04)	2.4	4.5
TOMPKINS-BECKWITH	(0.69)	(0.81)	(0.12)	19.7	36.2
SUB-TOTAL	(2.20)	(2.49)	(0.29)	48.7	

NON-CRITICAL CONTRACTS

	DEC.	JAN.	△	WT. FACT.	% COMPLETE
J. A. JONES	(0.32)	(0.32)	0.00	26.6	97.2
OUTLYING BLDGS.	(0.35)	(0.35)	0.00	0.6	16.9
TELLEPSEN	(0.53)	(0.30)	(0.27)	2.4	67.6
BOH BROS.	0.00	0.00	0.00	2.4	95.0
GULF ENG.	(0.02)	(0.03)	(0.01)	1.2	59.0
WESTINGHOUSE	0.00	(0.12)	(0.12)	1.7	16.3
FEGLES	0.00	(0.06)	(0.06)	.5	0.00
SUB-TOTAL	(1.22)	(1.68)	(0.46)	33.0	
TOTAL	(3.42)	(4.17)	(0.75)		



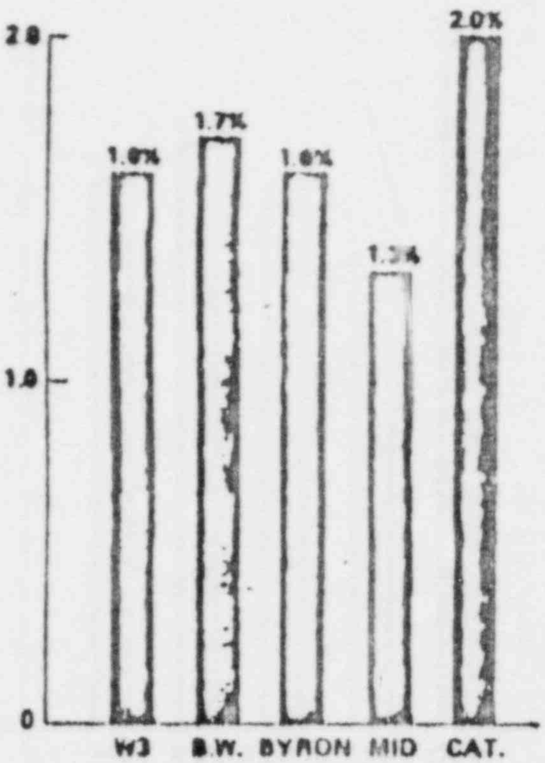
RATE PER MONTH NEEDED
TO ACHIEVE SCHEDULED
FUEL LOAD



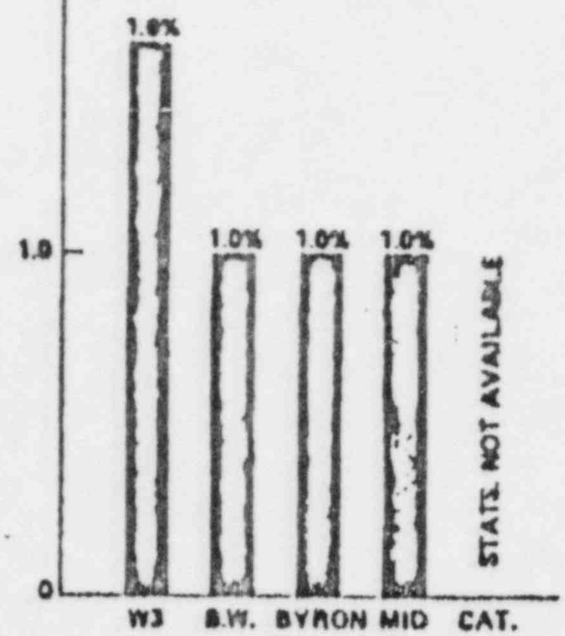
AVERAGE RATE PER MONTH
FOR QUARTER
9/30/78 → 12/31/78

APR 5 1979

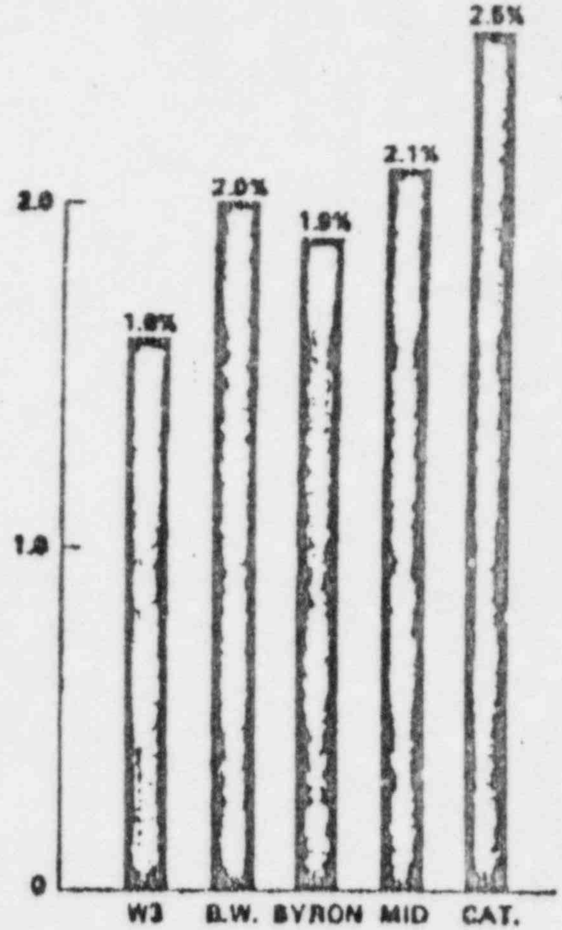
PWR'S SCHEDULED FOR FUEL LOAD 1979 - 1982 THAT ARE BETWEEN 40 & 60% COMPLETE



ACTUAL RATE PER MONTH
SINCE JULY, 1978



ACTUAL RATE PER MONTH FOR
LAST QUARTER (NOV. 1978–FEB. 1979
YELLOW BOOKS)



AVERAGE RATE PER MONTH
REQUIRED TO ACHIEVE
SCHEDULED CORE LOAD

W3 - WATERFORD 3
BW - BRAIDWOOD 1
BYRON - BYRON 1
MID - MIDLAND 1

CAT - CATAWBA 1

ENCLOSURE 4

APR 5 1979

RESPONSES WHICH WE WOULD
LIKE EARLY FEEDBACK ON

QUESTION #	TOPIC	PROBLEM
22.3	DIVERSE CONTAINMENT ISOLATION	a) ACCEPTABILITY OF ANSWER b) CONSTRUCTION STATUS
22.8	CONTAINMENT PURGING	ACCEPTABILITY OF TECH SPEC
22.10	THERMAL ANALYSES (MSLB) OF SELECTED COMPONENTS	EXCEPTIONS TO CSB INTERIM MODEL
40.17	ELECTRIC CABLE COLOR CODING	a) ACCEPTABILITY OF ANSWER b) CONSTRUCTION STATUS
110.1	NORTH ANNA	a) TIME b) MONEY c) NEED d) DESIGN OF FUEL e) APPENDIX K
211.12	SAFETY SEQUENCE LOGIC DIAGRAMS FOR C #15	a) TIME b) MONEY c) NEED TO "COMPLEMENT" FSAR
313.11	PROTECTION AGAINST CL_2 AND NH_3	ACCEPTABILITY OF DESIGN CRITERIA

Southern California Edison Company

P. O. BOX 800

2244 WALNUT GROVE AVENUE

ROSEMEAD, CALIFORNIA 91770

J. H. DRAKE
VICE PRESIDENT

TELEPHONE
213-572-2258

SCE

August 16, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

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

Following the events at Three Mile Island, Southern California Edison Company and San Diego Gas & Electric Company launched an evaluation program directed toward identifying appropriate changes to the design and operation of our San Onofre nuclear facilities. This program is comprehensive, including evaluations by ourselves, our contractors, and several consultants, as well as a review of all recommendations put forth by industry and governmental agencies.

The program was undertaken with the primary objective of defining, at the earliest possible date, all design and operational changes deemed necessary to modify San Onofre in a manner consistent with the latest consensus of what constitutes an upgraded level of nuclear safety while maintaining our operating license schedule.

The attached report entitled, "Evaluation and Action Plan for San Onofre Nuclear Generating Station Units 2 and 3 Relative to the Three Mile Island Incident," dated August, 1979, presents the results to date of our continuing evaluation. The report provides commitments to proceed with the items identified in the Lessons Learned Task Force (with the exception of those items that require rulemaking) as well as additional items that we have identified as desirable to provide an upgraded level of nuclear safety. We have initiated engineering and procurement efforts required to implement the commitments contained in the report.

~~8210290001~~3
PDR/LPDR

SCE

In a meeting today, your staff was briefed on the additional steps we have taken to complete physical construction of Unit 2 by November, 1980, and the fact that we have submitted the information requested by your staff on the majority of the open items in our operating license review docket.

Accordingly, there is now a satisfactory basis for the staff to proceed with review of our operating license application on a schedule consistent with receipt of an operating license by November, 1980. Aside from the fact that timely operation of Unit 2 will result in the saving of approximately 925,000 BBL of oil per month and $21,000,000 per month to our customers in fuel costs, this date is extremely important to Edison and San Diego because of our projected system demand requirements in 1981.

While recognizing that material availability and other factors outside of everyone's control, as discussed with your staff, pose an unprecedented challenge to complete TMI-related work in accordance with your desired regulatory schedules, it is nonetheless our intent to complete all work within the realm of physical possibility, and to work closely with your staff to maintain mutually acceptable schedules. Toward this end, if you consider it would be helpful, we are prepared to station some of our key technical personnel in our Washington, D.C. office for the purpose of uninterrupted liaison with your staff.

We will, of course, continue to follow the results of the Three Mile Island investigation and revise our plans as a result of additional changes that may be deemed necessary.

Because we are proceeding to have San Onofre, Unit 2 complete and ready for fuel loading by November, 1980, it is respectfully requested that your staff review of our operating license application be placed back on schedule for issuance of a license by November, 1980.

Your prompt attention to this request will be greatly appreciated.

Sincerely,

J H Drake

Enclosure

EVALUATION AND ACTION PLAN FOR SAN ONOFRE NUCLEAR GENERATING STATION UNITS 2 & 3 RELATIVE TO THE THREE MILE ISLAND INCIDENT

AUGUST, 1979

SOUTHERN CALIFORNIA EDISON CO.
SAN DIEGO GAS & ELECTRIC CO.

~~8210290069~~
PDR/LPDR

EVALUATION AND ACTION PLAN

FOR SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2 & 3

RELATIVE TO THE THREE MILE ISLAND INCIDENT

AUGUST, 1979

Southern California Edison Co.

San Diego Gas & Electric Co.

EVALUATION AND ACTION PLAN

FOR SAN ONOFRE NUCLEAR GENERATING STATION

UNITS 2&3

RELATIVE TO THE THREE MILE ISLAND INCIDENT

I. Introduction

The purpose of this report is to assist the Nuclear Regulatory Commission in its review of San Onofre Nuclear Generating Station Units 2&3 (SONGS 2 and 3) relative to the Three Mile Island (TMI) event. This report describes the results of Southern California Edison's (SCE's) extensive evaluation program concerning TMI and identifies appropriate actions for SONGS 2 and 3. The evaluation program included evaluations by SCE personnel, SCE contractors and several SCE consultants as well as a review of industry and government agency recommendations.

SCE considers that there are several design and operating features of SONGS 2 and 3 that minimize the probability of a TMI event at SONGS 2 and 3. Nevertheless, SCE has identified design and operational changes which will render that probability even more remote. SCE is proceeding with the engineering and procurement required to support these changes with the intent of completing them prior to fuel load. The ability to achieve this goal is dependent on procurement schedules, available technology, access to NSSS resources for design and analysis, and other items beyond SCE's control.

In addition, SCE is proceeding with numerous evaluations to determine the desirability of additional design changes. SCE will continue to monitor the activities of industry and regulatory review groups to identify additional items which should be considered in the design and operation of SONGS 2 and 3.

II. Action Items Planned To Be Accomplished Prior To Fuel Load

A. Plant Design

II.A.1 Action Item: T_H Subcooling Indication

Schedule: Prior to Fuel Load

Background:

Certain operator actions were taken during the TMI scenario because the operators were unaware that the reactor coolant was in a saturated condition. The formation of voids in the primary coolant as a result of flashing was a contributing factor in the loss of natural circulation.

Approach:

The SONGS 2 and 3 instrumentation design is being reviewed to establish an appropriate method to alert the operator when the reactor coolant system is approaching conditions of saturation. The method to be implemented is expected to include the installation of a system with software that provides the operator with an on-line indication of the margin to coolant saturation conditions and will include procedures to correlate this with the other related plant parameters.

The condition of the reactor coolant system should also be closely monitored during heatup and cooldown operations to assure that (a) pressure-temperature specifications regarding saturation are maintained, and (b) an unexpected excursion does not result in saturation conditions which could affect natural circulation. SCE will verify the applicability of the effective ranges and accuracies of the instruments selected above, including high temprature measurement capability.

II.A.2 Action Item: Reactor Vessel Level Indication

Schedule: Prior to Fuel Load

Background:

During the sequence of events at TMI, the fuel was left exposed for a period of time without operator knowledge of that condition. The operators had no direct indication of water level in the reactor vessel. Vessel level was inferred by measuring the water level in the pressurizer; however, this instrumentation did not provide accurate information due to the saturated condition of the coolant, and later, due to the presence of the hydrogen bubble in the vessel head.

Approach:

The SONGS 2 and 3 reactor instrumentation is being reviewed with the objective of identifying diverse ways to measure reactor vessel water level. SCE will implement one or a combination of the following to indirectly indicate vessel level:

(a) core thermocouples with wide temperature range

(b) ex-core and in-core detector signals

(c) coolant inventory control procedures

A direct reactor vessel level indication system is also being evaluated and will be implemented. The system may be based on differential pressure measurements made at preselected points above the top of the fuel, or on conductivity probes, heated thermocouples, ultrasonics or gamma and neutron void detectors. The system design for direct level indication will be selected based on equipment availability and the outcome of the investigation of the above diverse methods.

II.A.3 Action Item: Broad Range Instrumentation

Schedule: Prior to Fuel Load

Background:

During the course of events at TMI, particularly in the post accident period, much information on reactor coolant system parameters was not available because the instrument range was inadequate. In retrospect, several primary and secondary system parameters can be identified as desirable information from both an action and a diagnostic viewpoint.

Approach:

SCE will investigate and identify those instrument channels that are desirable to support emergency procedures and the ranges over which each parameter should be monitored. The worst case environment in which each channel should be operable will also be identified, and based on this, instrumentation will be upgraded or added. Several parameters where the conventional range is expected to be increased are hot and cold leg temperatures, in-core thermocouples, and temperature channels for secondary liquid and steam systems.

II.A.4 Action Item: Pressurizer Level Indication Upgrade

Sche..le: Prior to Fuel Load

Background:

During the TMI transient, the reactor coolant system conditions of temperature and pressure were such that the pressurizer level indication did not provide the information necessary to know the actual level of coolant in the reactor vessel and piping. In addition, the voiding and the wide temperature excursions may have affected the accuracy of the indicated level in the pressurizer.

Approach:

The differential pressure level indicators used on the SONGS 2 and 3 pressurizers are calibrated based on temperature compensation to account for the large difference between the hot leg and the reference leg. SCE will investigate the usefulness of level indication using either a wider range of temperature compensation or a dynamic compensation.

II.A.5 Action Item: Containment Isolation Upgrade

Schedule: Prior to Fuel Load

Background:

For transients with low mass release rates to the containment, significant time can elapse before a containment isolation actuation signal (CIAS) is generated on containment high pressure. CIAS on direct indication of potential radiological release from containment could significantly reduce the potential release of radioactivity under these conditions.

Approach:

The SONGS 2 and 3 containment isolation system will be modified to provide a CIAS on containment high pressure (4 psig) or actuation of the Safety Injection System. An evaluation will be conducted to determine which in-containment systems would be desirable under various event scenarios and methods and procedures for allowing continued operation of these systems following a CIAS will be evaluated and implemented where necessary. Provisions will be implemented to eliminate the potential for inadvertent reopening of isolated lines when CIAS is reset.

II.A.6 Action Item: Emergency Power Supply for Pressurizer Heaters

Schedule: Prior to Fuel Load

Background:

As a result of an examination of the TMI events, it became apparent that the number of challenges to the emergency core cooling systems should be minimized. In this regard, events involving loss of offsite power should be particularly scrutinized to assure that control is not lost or impaired to those systems intended as the primary means to control pressure or cool down the RCS. Two systems in this category are pressurizer backup heaters and pressurizer level indication. These are required to maintain control of natural circulation following unit trip. At SONGS 2 and 3, redundant channels of pressurizer level indication are displayed in the control room and at the Remote Shutdown Panel (RSP) and are powered from a 1E instrument bus.

Approach:

SONGS 2 and 3 each have redundant pressurizer backup heaters powered from Class 1E diesel buses. They can be manually controlled by the operator from either the control room or the RSP. SCE will conduct an evaluation to verify the adequacy of the existing heater capacity to control primary plant pressure during natural circulation following unit trip and in maintaining a hot-standby condition; modifications resulting from this evaluation will be implemented.

II.A.7 Action Item: Direct Indication of Safety Valve Position

Schedule: Prior to Fuel Load

Background:

A major contributing factor to the TMI event was the failure to close of a power operated relief valve (PORV) and the subsequent failure of the operators to rapidly recognize that the PORV had stuck open. SONGS 2 and 3 do not utilize PORV's. Overpressurization is prevented by spring loaded safety valves installed on the pressurizer (PSV's) which are designed to lift at a pressure considerably above that at which the TMI PORV's were set to lift.

At present, SONGS 2 and 3 utilize temperature indication immediately downstream of the PSV's and various types of instrumentation on the quench tank (into which the PSV's discharge) to detect flow out of the valves.

Approach:

Based on the importance attributed to the operator knowing that the PSV has failed open and in order to avoid total dependence on temperature or level instrumentation downstream of the PSV's, a direct indication of either valve position or flow through the valve will be incorporated. This indication will be displayed in the control room.

II.A.8 Action Item: Non-Isolation of Reactor Coolant Pump (RCP) Support Systems

Schedule: Prior to Fuel Load

Background:

Current RCP support systems are isolated on a containment isolation actuation signal (CIAS) and/or on a safety injection actuation signal (SIAS). Accident scenarios typically assume that the RCP's trip and are not considered for post-accident control.

However, during some accident conditions, operation of one or more RCP's may be desirable to augment core cooling capability. Mitigation of some small break LOCA's may be enhanced by RCP availability.

Approach:

In conjunction with item II.A.5, the support systems for the RCP's will be evaluated and modifications identified and implemented which would allow the operator to have the capability to resume or continue RCP operation.

II.A.9 Action Item: Control of Containment Sump Pump Operation

Schedule: Prior to Fuel Load

Background:

As a result of overfilling the RCS quench tank at TMI, water accumulated in the containment sump. The containment sump pumps were apparently not designed for either (1) manual operation or (2) automatic isolation. This resulted in the overflow of radwaste storage tanks and release of radioactivity to the environment.

Approach:

The SONGS 2 and 3 design is being reviewed with respect to sump pump operation and isolation. One or both of the following changes will be implemented:

1. Sump pump operation changed to manual control.

2. Sump pump control changed such that isolation would occur upon CIAS or SIAS.

II.A.10 Action Item: Plant Shielding in Areas Occupied During Post-Accident Operations

Schedule: Prior to Fuel Load

Background:

Following an accident, radiation levels in the vicinity of certain tanks, filters, ion exchangers and piping systems can increase by several orders of magnitude above normal levels, and remain at high levels for long periods of time. In addition, systems that were not designed to contain large radiation sources may become highly radioactive. Occupational exposures will result primarily from post-accident manual operations. Such operations include valve line-ups, sampling, recovery operations, equipment repair or removal, and resin and filter changeouts.

Approach:

A design review is being performed on the radiation rate and the effective shielding in areas around systems that may, as a result of an accident, contain highly radioactive materials. Major areas of concern would be the auxiliary building, radwaste control stations, emergency power supplies, motor control centers, and instrument areas.

After the design review is complete, provisions shall be made such that there will be adequate access to vital areas and protection of safety equipment. These provisions will include design changes, increased permanent or temporary shielding and/or post-accident procedural controls.

II.A.11 Action Item: Auxiliary Feedwater Flow Indication

Schedule: Prior to Fuel Load

Background:

The inability of the auxiliary feedwater system (AFWS) at TMI to respond when required (even though due to a maintenance error) illustrated the importance of AFW to the safe operation of the plant under various transient and accident conditions. The SONGS 2 and 3 AFWS has been reviewed in detail and the design and function of the system has been verified to meet the following requirements:

(1) single failure criteria

(2) testabiity of initiating signals

(3) all operating and control power from emergency buses

(4) manual initiation from control room with no bypass of safety function

(5) automatic actuation and sequencing when required

In addition, a single channel of safety grade indication is provided in the control room to measure auxiliary feedwater flow to each steam generator.

Approach:

An evaluation of the adequacy of the existing single channel of auxiliary feedwater flow in conjunction with steam generator level indication to satisfy power diversity requirements will be made. If such adequacy cannot be established, an additional channel of safety grade instrumentation will be installed.

II.A.12 Action Item: Improve In-plant Iodine Instrumentation

Schedule: Prior to Fuel Load

Background:

During the TMI accident, radioactive water was transferred from the containment to the auxiliary building. Radioactive gas evolved from this water and entered the auxiliary building atmosphere. Subsequently, some airborne activity entered the control room. The methodology used in determining the radioactive iodine concentration produced results which greatly overestimated the iodine concentration. As a result, plant personnel were required to perform operations functions while using respiratory equipment. This sharply limited their communication capability and diminished personnel performance during the accident.

Approach:

Equipment and/or instrumentation will be provided for more accurately determining in-plant airborne radioactive iodine concentrations under accident conditions. This will minimize the unnecessary use of respiratory protection equipment.

II.A.13 Action Item: Onsite Technical Support Center

Schedule: Prior to Fuel Load

Background:

The conduct of on-site operations at TMI subsequent to the initiation of the event and particularly during the post-accident phase pointed out the need to improve the procedures, information availability and execution process for the actions necessary to mitigate the accident condition.

Approach:

An Onsite Technical Support Center (OTSC) will be established at SONGS 2 and 3. The OTSC will be established based on consideration of the following criteria:

(1) separate from and in close proximity to the control room

(2) capability to display and transmit information pertinent to plant status

(3) post-accident habitability

(4) capability to serve as implementation center for emergency plans

(5) access to the plant as-built documentation

II.A.14 Action Item: Onsite Operational Support Center

Schedule: Prior to Fuel Load

Background:

The incident at TMI indicated a need for a centrally located operational assistance area (separate from the control room) to aid in plant operation.

Approach:

An Onsite Operational Support Center (OOSC) will be established at SONGS 2 and 3. The OOSC will serve as a "holding area" outside of the control room for shift support personnel (other than those in the control room) to report to and receive assignments. The OOSC will have communication capability with the control room.

II.A.15 Action Item: Pressurizer Safety Valve (PSV) Two Phase and Subcooled Liquid Flow Verification

Schedule: Prior to Fuel Load

Background:

PSV's are designed and tested primarily for steam flow. However, the events at TMI have shown that the potential for two phase or subcooled liquid flow exists; this presents possible uncertainties regarding the capability of such valves to structurally withstand the effects of two phase or subcooled liquid flow, and to reclose following such flow conditions.

Approach:

SCE will develop and submit to the NRC, a testing program to confirm the ability of the SONGS 2 and 3 PSV's to structurally withstand the impacts of, and reseat against, two phase and subcooled liquid flow.

The program will be completed by July, 1981.

II.A.16 Action Item: Analysis of Design and Off-Normal Transients and Accidents

Schedule: Prior to Fuel Load

Background:

The TMI accident indicates that further analyses are needed to determine system behavior following small break LOCAs.

The response of the primary system to a small break LOCA varies depending on the break size, the location of the break, and availability of reactor coolant pumps, safety systems, and secondary side cooling. Further analyses of the other transients and accidents are also needed to (1) identify proper operator actions in response to transients, (2) develop guidelines for emergency procedures and (3) implement operator training.

Approach:

Small break LOCA analyses will be performed to determine the system response from the initiation of the small break LOCA to a stable shutdown condition. These analyses will utilize, to the extent applicable, generic CE small break LOCA analyses. The analyses will determine the effect on system behavior of continued reactor coolant pump operation, the effect of loss of offsite power, and the effect of losing steam generator cooling capability to assure core coolability in all cases and adequacy of operator response and emergency procedures. The analyses will address conditions of inadequate core cooling in order to identify operator actions in recognizing the symptoms of inadequate core cooling and appropriate corrective action.

Analysis of other transients and accidents will be performed, to the extent necessary to verify the applicability of generic CE transients and accident analyses results to provide input to the development of guidelines for operator response to these events. These analyses will be performed using realistic input assumptions and best estimate modeling. Event tree analysis methodology will be used to determine combinations of initiating events and additional consequential equipment failure or operator response error.

II.A.17 Action Item: Anticipatory Trip on Loss of Feedwater

Schedule: Prior to Fuel Load

Background:

At TMI, the lack of direct initiation of reactor trip on the loss of feedwater initiating event caused the primary relief valve to open and its subsequent failure to close contributed directly to the severity of the event. Complete loss of feedwater events in operating CE plants initiate a low steam generator water level reactor trip in approximately 25 seconds. Typically, 15 minutes is available to initiate auxiliary feedwater flow to the steam generator prior to dryout. An anticipatory trip would extend this time, to reduce the possibility of a complete loss of secondary heat sink. Such a trip would reduce the severity of any potential overpressure condition on the primary side and decrease the probability of challenges to the safety valves.

Approach:

The extension of steam generator dryout time by incorporating an anticipatory reactor trip on loss of feedwater will be evaluated and design changes implemented.

II.A.18 Action Item: Remotely Operated Reactor Vessel Vent (Safety Grade)

Schedule: Prior to Fuel Load

Background:

During the TMI accident, fuel cladding overheated, reacted with the reactor coolant water, and thereby generated a large quantity of non-condensible gases which ultimately adversely affected core cooling. These gases accumulated at the top of the reactor vessel (and possibly, in other high points of the RCS) and hampered forced circulation and the ability of the RCS to sustain natural circulation.

Approach:

RCS venting and degassing procedures have been established for normal plant operation at SONGS 2 and 3. SCE will review the current SONGS 2 and 3 design in order to establish a configuration which will provide the capability to vent the reactor vessel. The evaluation of such venting capability will include investigating remote manual operation from the control room.

II. Items Planned To Be Accomplished Prior To Fuel Load

B. Plant Operations

II.B.1 Action Item: Improved Operator Training

Schedule: Prior to Fuel Load

Background:

At TMI, several actions were taken which led to more serious consequences and other actions which would have mitigated the effects of the loss-of-coolant were not taken. Although some of the important information available to the operator was contradictory, an awareness of the reactor coolant system condition which would exist during small break LOCA's and the key instrumentation to diagnose those conditions would have had considerable value.

Additionally, and although not directly related to the diagnosis of the loss-of-coolant, TMI operators expended a significant effort to diagnose and correct the misaligned auxiliary feedwater system in the ten minutes immediately following the unit trip. Had this safety system been properly aligned, a more orderly condition in the control room would have existed during this critical time period.

Approach

To assure a more thorough understanding of design and off-normal events and the manner in which they may be influenced by operator action, additional operator training is being implemented. The areas of training include:

(1) Transients and accidents specifically related to the results of the analyses described in section II.A.16 of this report.

(2) A review of the TMI scenario including the following aspects of the event:

  (a) cause of the loss of reactor fluid.

  (b) alignment of the auxiliary feedwater system.

  (c) instrumentation indications which are available for diagnosis of the loss of coolant.

  (d) the release of radioactive fluid from containment.

  (e) high pressure safety injection system operation during the event.

(3) Diagnosis and mitigation of small break LOCA's.

(4) Control of RCS cooldown by natural circulation.

(5) Importance of maintaining RCS fluid temperature below saturation.

II.B.2 Action Item: Improved Emergency Procedures

Schedule: Prior to Fuel Load

Background:

Inadequate coordination of transient and accident analyses with operating procedures contributed to inappropriate operator action at TMI.

Approach:

Emergency procedures will be prepared which reflect appropriate consideration of the results of the analyses described in section II.A.16 of this report. Specifically, the procedures will be directed toward:

(1) improving operator performance during small break LOCA's

(2) assuring that the operator can recognize and respond to conditions of inadequate core cooling

(3) improving operator performance during transients and accidents, including events that are caused or worsened by operator action.

II.B.3 Action Item: Shift Supervisor Responsibilities

Schedule: Prior to Fuel Load

Background:

Regulatory review of the TMI accident has suggested a need for more definitive specification of reactor operations command and control responsibilities and authorities.

Approach:

SCE will review its current procedures and policies regarding reactor operations command and control reponsibilities and authorities including those during accident conditions to ensure responsibilities, duties, authority, and lines of command are clearly defined. Revisions shown to be appropriate by this review will be implemented.

II.B.4 Action Item: Shift Turnover Improvement

Schedule: Prior to Fuel Load

Background:

The operational occurrences at TMI indicate a need for a more formal shift turnover to ensure the current plant conditions are made known to the oncoming shift and to ensure proper turnover of command operations.

Approach:

SCE will review its shift turnover procedures and implement any changes necessary to ensure oncoming shift personnel are aware of plant conditions. Additionally, SCE will utilize supplementary checklists for instrument technicians, maintenance personnel, and auxiliary operators. A means to evaluate the effectiveness of such turnover procedures will be established.

II.B.5 Action Item: Control Room Access Control

Schedule: Prior to Fuel Load

Background:

An excessive number of people in the control room during even normal conditions can lead to operational errors and confusion.

Approach:

More stringent procedures with respect to personnel access to the control room during normal and abnormal conditions will be evaluated and implemented.

II.B.6 Action Item: Systems Integrity for High Radioactivity

Schedule: Prior to Fuel Load

Background:

At TMI, some systems external to the containment building that carried radioactive liquid leaked. There was apparently little prior knowledge of their operational leakage characteristics. Such leakage complicated the management of radiation control activities.

Approach:

Leak rate testing will be performed on those systems outside of containment that could contain high level radioactive materials following accident conditions. Additionally, SCE will implement a periodic testing and maintenance program to minimize such leakages.

II.B.7 Action Item: Emergency Planning Improvements

Schedule: Prior to Fuel Load

Background:

An effective response to an emergency involving offsite consequences is dependent upon adequate preplanning and timely and accurate communication during the emergency. The TMI accident suggests the need for increased level of preplanning and communications.

The Emergency Plan for San Onofre was prepared incorporating the Emergency Response Plans for all potentially affected offsite agencies and with their cooperation. The Emergency Plan addreses notification procedures, the activities of each agency involved, potential radiation exposure criteria for recommending evacuation and possible evacuation routes. The Emergency Plan also requires periodic drills and periodic review of the procedures to ensure tnat the plans are current and adequate.

Approach:

The Emergency Plan is currently being reviewed in light of TMI to ensure that its provisions are adequate to deal with an occurrence such as TMI. Improvements which are identified by this review will be implemented.

II.B.8 Action Item: Provisions for Notification of NRC

Schedule: Prior to Fuel Load

Background:

The NRC should be notified as soon as possible once it is determined that the reactor is not in a controlled or expected condition of operation, and open communication should be maintained for the duration of the event.

Approach:

San Onofre 2&3 Station emergency procedures will be revised to include provisions for notifying and maintaining communication with the NRC within 1 hour of occurrence of a situation involving uncontrolled reactor operation.

II.B.9 Action Item: Onsite Shift Technial Advisor

Schedule: Prior to Fuel Load

Background:

Regulatory review of the TMI accident has suggested the desirability of complementing the control manipulation, event diagnosis and operations management functions of shift operations personnel with an individual capable of providing professionally qualified technical support.

Approach:

SCE will provide a qualified engineer (i.e. Bachelor's degree or equivalent in an engineering discipline) on shift on the site to serve in an advisory capacity to shift supervisors. Such an advisor will have knowledge of plant response to transient and abnormal occurrences.

III. Action Items Planned to be Accomplished as soon as Practicable

A. Plant Design

III.A.1 Action Item: Containment Water Level Indication Upgrade

Schedule: As Soon As Practicable

Background:

During the TMI accident, the pressure relief quench tank overflowed onto the containment floor. The water, due to the stuck open relief valve, was of sufficient volume to rise above the range of the containment water level indicator. Therefore, the flood level inside containment was not known. It is desirable to have indication of containment water level in order to know what equipment or instrumentation is being flooded. The San Onc.re containment water level indicator is capable of measuring approximately 92" of water above the sump minimum level. This height corresponds to approximately 2 feet above the containment floor. The maximum credible flood level inside containment is approximately 7 feet above the floor level. Therefore, it is possible that level indication inside containment could be lost.

Approach:

The design of the containment level indicating system will be modified as necessary to provide level indication to the maximum credible flood level.

III.A.2 Action Item: Improve Post-Accident Sampling Capability

Schedule: As Soon As Practicable

Background:

Following an accident in which fuel damage occurs, the primary system water and the water and air in containment may be highly radioactive. chemical and radiological analysis would be performed to provide information on radioactivity levels and other information necessary for the operators to understand the post-accident plant conditions and to help determine subsequent actions to maintain the plant in a safe condition.

High radiation levels can inhibit the ability to obtain the needed samples and perform the necessary analysis. Also high background radiation and/or high sample radiation may render the in-plant radiological spectrum analysis equipment inoperable during and after an accident.

Approach:

A design review is being performed on the reactor coolant and containment atmosphere sampling system to determine the capability of personnel to obtain a sample under accident conditions without incurring a radiation exposure to any individual in excess of 3 Rems whole body and 18 3/4 Rems to the extremeties. After the design review is completed, any additional design features and/or additional shielding that are required to insure that personnel will be able to promptly and safely obtain the necessary samples will be completed.

In addition, a design review is being performed on the radiological spectrum analysis facility to determine its capability to analyze in a post-accident environment for the radioisotopes which are the indicators of the degree of core damage. Design or procedural changes indicated by the review will be implemented.

III.A.3 Action Item: Increase Range of the Radiation Monitors

Schedule: As Soon As Practicable

Background:

Effluent radiation monitors are designed to detect and measure releases associated with normal reactor operations and anticipated operational occurrences. These monitors do not have sufficient range to function under release conditions associated with certain accidents.

The containment radiation monitoring instrumentation is used in assessing plant conditions during and following an accident. The radiation level inside containment after certain postulated accidents would be greater than the level that containment radiation monitors are typically capable of monitoring.

Approach:

The following actions will be completed to increase the range of the radiation monitors for SONGS 2 and 3:

(1) A noble gas effluent monitoring system with an upper detection range capability of 10^5 u Ci/cc (Xe-133) will be installed.

(2) One of the following items will be completed:

(a) The adequacy of the Emergency Radiation Monitoring System (ERMS), which is located outside containment, in measuring radiation levels up to 10^8 R/hr inside containment will be established, or

(b) Two separate radiation level monitors with a maximum range of 10^8 R/hr will be installed inside containment. These monitors will be qualified to the design criteria for safety-grade instrumentation.

(3) A means of monitoring radioactive iodine in effluents during accident conditions will be provided. The sampling will be conducted by adsorption on a media.

IV. Additional Action Items To Be Initiated

A. Plant Design

IV.A.1 Action Item: Reactor Coolant Pump (RCP) Upgrading

Background:

The operators at TMI stopped the RCP's to preclude possible pump damage from operation at the time that considerable boiling existed in the RCS. Stopping the RCPs and not being able to initiate natural circulation resulted in no flow conditions and contributed to further fuel damage.

Approach:

The potential benefits/implications of upgrading and modifying the RCP's for post-accident operation to enhance mitigation of some LOCA's will be evaluated.

IV.A.2 Action Item: Control Room Display Improvements

Background:

Some reviews of the TMI accident have indicated that improvements could be made to the display of plant status information in the control room.

Approach:

An evaluation will be made of the current SONGS 2 and 3 control room design and the plant computer system. This review will take into consideration the signals recorded by the computer and the form of data presentation. It will also consider location and layout of information and control functions, the form and content of operating instructions and will stress the suitability of these functions for the decision making process under normal and emergency conditions.

IV.A.3 Action Item: Post-Accident Waste Processing Methods

Background:

The introduction of abnormally high activity liquid and gaseous waste to existing radioactive waste processing systems would result in a decrease in equipment effectiveness and in premature equipment failure. This would impair the ability of the various systems to process the waste.

Approach:

SONGS 2 and 3 design will be evaluated to determine whether temporary or permanent post-accident clean up systems could be put in place following an accident. The following will be considered:

(1) Containment penetrations to allow for installation of containment gas clean up, including additional hydrogen recombiners and a charcoal delay or cryogenic system.

(2) The capability to install portable, skid-mounted high level waste processing system for liquid, gaseous and solid waste.

(3) The capability to install an alternate system for the draining of the containment sumps.

(4) The capability to install additional equipment for maintaining the proper waste chemistry thus reducing the chances for equipment malfunction or failure.

In addition, an evaluation of the effects of decontamination chemicals on the Radwaste System to insure that these chemicals will not adversely impact system performance will be performed.

IV.A.4 Action Item: Instrument Sensor Location

Background:

During the TMI accident, the containment water level rose above the level of certain equipment and instrument sensors.

The location of various sensors inside containment which would be desirable post-accident should be evaluated so that sensors not qualified for submersion would be above the high water level.

Approach:

SCE will continue to evaluate the instrumentation which is desirable following an accident. The evaluation and qualification of those sensors will be evaluated with respect to maximum containment water level and whether the sensor will function submerged.

IV.A.5 Action Item: Post-Accident Waste Holdup and Processing Capacity

Background:

The existing waste handling sytems at TMI were not designed to process the quantity or the radioactivity of the waste generated following the accident.

Approach:

The SONGS 2 and 3 waste systems will be reviewed to determine if an increase in the waste handling systems' flexibiity and overall processing capability could be accomplished following an accident. Areas to be evaluated include:

1. The feasibility of cross-connecting the clean and dirty liquid waste processing holdup and processing capacity.

2. The feasibility of additional connections from non-radioactivve waste systems to the radioactive waste systems. (Currently SONGS 2 and 3 turbine building sumps can be diverted to radwaste.)

3. The feasibility of adding process piping to allow for the recirculation of waste back into the containment for decay.

4. The feasibility of installation of a hydrogen recombiner to the Gaseous Waste System as a means to increase the effective storage volume and allow for a longer decay time.

5. The feasibility of connecting the vents from all liquid waste tanks to the Gas Waste System so as to allow for the processing of this gas prior to release.

6. The feasibility of adding remote indication and control instrumentation to allow for the processing of higher activity waste without increasing personnel exposure.

IV.A.6 Action Item: Direct Valve Position Indication

Background:

Position indication is used for verification and surveillance to ascertain initial valve lineups or to alert an operator that a valve is out of service or in an abnormal position. It can also be used as a control tool to monitor flow in a fluid or steam system. In most cases, a limit switch is used as a direct method to verify valve position and important systems are generally indicated in the control room. At TMI, the procedures used to assure auxiliary feedwater valve availability were insufficient.

Approach:

SONGS 2 and 3 presently have active components in the Engineered Safety Features (ESF) and auxiliary feedwater systems linked to a "Bypass-Inoperable Status Panel" which indicates to the operator in the control room that there is power to each component. Each redundant train is separately alarmed. SCE is evaluating several aspects of this system for improvement including: (a) providing a separate alarm if both redundant trains are inoperable; (b) providing positive indication for all valves in a safety train that can affect operability (this would include direct position indication using limit switches); and (c) reviewing valve control logic schemes for erroneous information indication under cerain failure modes.

IV.A.7 Action Item: Auxiliary Feedwater Actuation on Loss of Feedwater

Background:

At TMI, auxiliary feedwater valves were designed to open on steam generator low water level within approximately 40 seconds following loss of feedwater. Auxiliary feedwater was not admitted because the auxiliary feedwater block valves were shut. The automatic initiation of auxiliary feedwater on loss of feedwater in addition to steam generator water level would provide earlier actuation of auxiliary feedwater for loss of feedwater transients, and maintain a greater inventory in the steam generator to enhance RCS cooling during these transients.

Approach:

The SONGS 2 and 3 design presently provides for initiation of auxiliary feedwater on low steam generator level. In addition, the actuation of auxiliary feedwater on complete loss of main feedwater will be evaluated to determine its effect on normal auxiliary feedwater reliability and system response to a loss of feedwater transient.

IV.A.8 Action Item: Post-Accident Effluent Releases

Background:

During the course of the accident at TMI, some radioactive releases occurred. Preliminary analysis indicates that there were two principal pathways of release; the first was evolution of gases from the water transferred into the auxiliary building; the second was the release of gases from the hold up tanks and as a result of filter handling operations.

Approach:

The following aspects of conceivable unplanned or uncontrolled releases at SONGS 2 and 3 will be evaluated:

1. The routing of all vents and drains from equipment which might conceivably handle post-accident fluids.

2. Guidelines for controlled release of all plant effluents.

IV.A.9 Action Item: Hydrogen Monitoring System

Background:

During the TMI accident, hydrogen gas was generated and released to the containment. Significant efforts were expended in attempts to quantify hydrogen levels within containment.

Approach:

Presently the SONGS 2 and 3 hydrogen monitoring system consists of two completely redundant trains and is designed to measure the hydrogen concentration inside the containment at two independent locations. The adequacy of this hydrogen monitoring system in light of the TMI accident will be evaluated.

IV.A.10 Action Item: Reactor Coolant Pump Trip

Background:

For a limited spectrum of small break LOCA's, continued operation of the Reactor Coolant Pumps following the break could result in a greater decrease in reactor coolant system inventory and thus aggravate the consequences of the event.

Approach:

Based upon the results of the analyses described in section II.A.16 of this report, SCE will evaluate the desirability of providing an automatic Reactor Coolant Pump trip following a specific range of small break LOCAs.

IV. Additional Action Items To Be Initiated

B. Plant Operations

IV.B.1 Action Item: Improved Operator Training

Background:

The items discussed in sectin II.B relate closely to the TMI sequence of events. Regulatory review of TMI has suggested the need to evaluate additional areas of operator training.

Approach:

The following potential changes to personnel training programs will be evaluated:

1. Formal training of non-licensed plant equipment operators (PEO's) and the requirement to pass an examination as a prerequisite to standing watch. This training would provide the PEO's with general nuclear power plant background and emphasize the importance of safety system requirements, maintenance/clearance procedures, and alignments in the area of the plant in which they are assigned responsibility.

2. Participation of plant support engineers in the operator training program.

3. Providing training for plant maintenance personnel and technicians in the area of clearance procedures.

4. Providing an additional full shift of operators to permit more time for retraining and a reduction in shift duty.

In addition, the following regulatory recommendations will be evaluated:

1. Formal establishment of minimum experience requirements for Senior Reactor Operator and Reactor Operator applicants.

2. Formal establishment of a requirement for instructors to hold Senior Operator Licenses.

3. Increasing minimum acceptable grades for plant administered (pre-licensing) examinations.

4. Increasing scope of operator training to include thermodynamics and hydraulics.

Appendix A
Comparison of San Onofre 2&3 Action Items
With NRC Recommendations Concerning
Three Mile Island

TMI-2 Lessons Learned Task Force Report Recommendations	San Onofre 2&3 Action Items
2.1.1	II.A.6
2.1.2	II.A.15
2.1.3.a	II.A.7
2.1.3.b	II.A.1, II.A.2 & II.A.3
2.1.4	II.A.5, II.A.8 & II.A.9
2.1.5	Not Applicable
2.1.6.a	II.B.6
2.1.6.b	II.A.10
2.1.7	II.A.11
2.1.8.a	III.A.2
2.1.8.b	III.A.3
2.1.8.c	II.A.12
2.1.9	II.A.16 & II.B.1
2.2.1.a	II.B.3
2.2.1.b	II.B.9
2.2.1.c	II.B.4
2.2.2.a	II.B.5
2.2.2.b	II.A.13
2.2.2.c	II.A.14
2.2.3	Not Applicable

I&E Bulletin 79-06B Items	
1	II.B.1
2	II.B.2
3	II.A.5
4	Not Applicable
5	Not Applicable
6	II.B.1
7	IV.A.6
8	IV.A.5
9	II.B.1
10	II.B.8
11	IV.A.10
12	Not Applicable

I&E Bulletin 79-06C Items	
1	Not Applicable
2	II.A.16
3	II.B.1
4	II.B.1
5	II.A.16 & II.B.1
Long Term Item 1	IV.A.11