Exhibit : I-2

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July 19, 1978

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Pennsylvania Public Utility Commission
Commonwealth of Pennsylvania
104 North Office Building
Harrisburg, Pennsylvania 17120

Attention: Honorable Louis J. Carter, Chairman

Subject: TMI-2 Start-up and Test Status

Dear Chairman Carter:

During the Met-Ed review on June 23rd, I reviewed the status of the TMI-2 start-up program and the schedule for commercial service. The purpose of this letter is to confirm our current assessment of the major problems that control the projected commercial service date of about November 1, 1978.

Start-up Program

A formal start-up testing program was planned and organized for Three Mile Island Nuclear Generating Station Unit 2 similar to the one that was conducted very successfully on Three Mile Island Nuclear Generating Station Unit 1. The program was scheduled for forty weeks as outlined on Attachment 1. The first event in the test program, hot functional testing, was completed in October 1977, in time to support the May 31, 1978 in-service date scheduled at that time. The test program is planned to fully exercise all of the plant equipment under carefully controlled and monitored conditions so that any deficiencies in design or construction can be identified. The end objective of the test program is to verify that the plant performs in full conformance with all operating and licensing specifications. While it is anticipated that some problems will occur during the test program, for administrative reasons the program schedule makes no explicit provision for delays.

By April 23rd, the test program had progressed to the "15-40 per cent power escalation" phase. In so doing the plant has operated at full temperature and pressure at a maximum electrical output of abou 200 Mw and has produced about 4000 Mwhrs. Attachment 2 is a list of some of the more significant problems which have arisen during the testing program. The last three items on Attachment 2 are particular worthy of discussion.

Control Assembly

In the TMI-2 reactor core each fuel assembly is fitted with either a moveable control rod, or a fixed burnable poison rod assembly (BPRA), or a fixed orifice rod assembly (ORA). The mechanism which locks the BPRA's and the ORA's into place showed wear at some other B&W installations. Consequently it was necessary to remove the reactor head and disassemble a portion of the reactor internals to correct this problem. That effort was completed during June concurrent with corrective action on the main steam safety valve problems.

Emergency Cooling

In March 1978 B&W notified us of an oversight in the safety analysis they had performed to verify adequate plant protection in the event of a loss of coolant accident caused by a small break in the reactor coolant system piping. The permanent solution to this problem involves the addition of piping and check valves to provide greater redundancy to the systems which provide emergency cooling. These changes may not be accomplished until the first refueling on TMI-2. There is a possibility that TMI-2 will be limited to about 93 per cent of full power pending the completion of the necessary modification. Met-Ed and GPUSC are pursuing an interim administratively controlled solution to remove the power restriction which has been accepted for TMI-1, and feel generally optimistic that it will be accepted for TMI-2.

Main Steam Relief Valves

On April 23, 1978, the reactor tripped while operating at 28 per cent power during the conduct of the start-up program. This type of transient leads to an increase in pressure in both the reactor plant and the steam plant. The pressure increase is controlled by main steam safety valves. The main steam safety valves open as a result of the increase in main steam pressure and relieve this pressure to the atmosphere; however, the main steam safety valves did not reclose when the pressure returned to its normal range. As a result of the safety valves failing to close appropriately, excessive heat was removed from the main steam system, cooling down the steam generators and thereby causing the reactor coolant system to cool down excessively. The rapid cooldown of the reactor coolant system, and the associated decrease in reactor coolant pressure, initiated injection of emergency cooling water in a manner similar to that expected during a loss of coolant accident.

During the course of this event, it was noted that liners from expansion joints in the discharge piping from the main steam safety valves had failed and were ejected into the air through the main steam safety valve discharge stacks.

Met-Ed and GPU Service Corporation established a Task Force to review and evaluate the causes and implications of this event and to recommend specific action to be taken to preclude such an occurrence in the future. It was well recognized by the Task Force and others that the main steam safety valves blew down excessively and while corrective action was necessary, it was believed at that time that normal adjustments to the valves would remedy that problem. The major thrust of the initial action was to correct the deficiencies in the discharge piping expansion joints, and such corrective action was completed about the middle of May. Concurrent with the repairs to the safety valve discharge piping, the plant was cooled down for cleanup of the chemicals added to the reactor coolant system in conjunction with injection of emergency cooling water and correction of other minor problems identified by the test program.

Upon return of the plant to normal operating temperature and pressure without nuclear power, main steam safety valve testing was initiated on May 18, 1978, with the expectation of adjusting the reclosure pressure of the valves to correct the excessive blowdown. It became apparent about May 20th that the allowable adjustments were not correcting the reclosure problem with the main steam safety valves.

On May 23, 1978, a meeting was held with engineering executives of the Lonergan Company (designers and manufacturers of the safety valves), and GPU Service Corporation, to determine the course of action that would be taken to correct the deficient valve operation. The Lonergan Company stated that two specific changes would result in acceptable valve performance. One change was a reduction of the back pressure caused by the design of the valve discharge piping and the second involved internal modifications to the valve. These changes were made on two valves and testing was resumed on May 26, 1978, with the modified discharge piping, and on May 31,1973, with the modified valves.

Concurrent with the joint effort with Lonergan, Burns and Roe (the Architect-Engineer for TMI-2) was directed to start engineering work to design modifications to the plant which would be necessary if the Lonergan valves had to be replaced. GPUSC personnel began immediately to canvass valve suppliers to identify the availability of replacement valves.

In the first part of June, a testing facility in Huntsville, Alabama became available for modification to permit off-site testing of the valves, and arrangements were made to test both modified and unmodified valves at that facility. In the meantime, testing at the plant continued through the 4th of June. Fifty-one (51) valve tests were accomplished in the period from May 18 through June 4, sixteen (16) of which were with modified valves. One hundred eleven (111) tests were conducted at Huntsville through June 22, 1978. None of the tests at Huntsville or at the plant site resulted in acceptable valve performance.

During the period of evaluation and testing from mid-May through June 23, many alternatives were considered for resolution of the problem. The search made throughout the United States for available replacement valves resulted in no valves being found that were available immediately that would fit the TMI-2 steam line configuration. The Forked River Nuclear Generating Station is being supplied valves of the same size as the Lonergan valves but which are made by another valve manufacturer but those valves will not be available until the end of November 1978. We were able to locate smaller valves that were available immediately and similar to the valves used for TMI-1.

On June 22, 1978, it was apparent that the valve testing at Huntsville was non-productive and that further testing of the Lonergan valves would not achieve satisfactory results. It was decided, therefore, to purchase twenty (20) smaller valves of the TMI-1 design to replace the twelve (12) Lonergan valves which did not perform adequately. By that time, the necessary modifications to the main steam lines had been identified and the necessary material to accomplish these modifications had been located, and procurement had commenced.

The steam line modifications involve welding into the four (4) main steam leads, a total of twenty (20) new nozzles. Welding the nozzles into place requires cutting holes in the main steam pipes, precision fit up of the new nozzles with the holes and completion of the welding process which must include in-process and post-weld inspections and post-weld heat treatment. In addition, the safety valve discharge piping inside the building has to be removed and replaced with different piping with a different configuration. All of this work must be accomplished in an elevated and congested area of the plant.

The question of why the plant was into start-up testing before the safety valve deficiency was identified deserves some comment. Probably the major contributor to that development was the excellent industry experience with steam safety valve performance. The situation is further complicated by the size of the valves. One of the valves will pass an amount of steam equivalent to that needed to generate 100 Mw of electricity and testing facilities for valves of that type are not generally available. The Huntsville facility did not become available until late 1976 and it is only marginally capable of testing these valves. The Lonergan valves were procured competitively and were the first of this size built by the company for the TMI-2 service conditions and were based upon an extrapolation of the design of a smaller valve with proven performance capability. This situation is not unusual as through the years the utility industry has frequently been forced to utilize equipment which could not be tested under operating conditions until completion of the plant construction.

July 19, 1978

Schedule

The main steam safety valve modification was initiated June 23,1978, and is expected to be complete about mid-August. Completion of the main steam safety valve modification in mid-August will permit a return to power early in September and an in-service date for the Unit at the end of October 1978, an overall delay of four (4) months. Attachments 3 and 4 show the main steam safety valve recovery schedule and the resulting overall schedule from start of testing to commercial operation assuming no further problems.

We will keep you informed of the TMI-2 start-up and test program status. If you or your staff require any further information, please call on us.

Very truly yours,

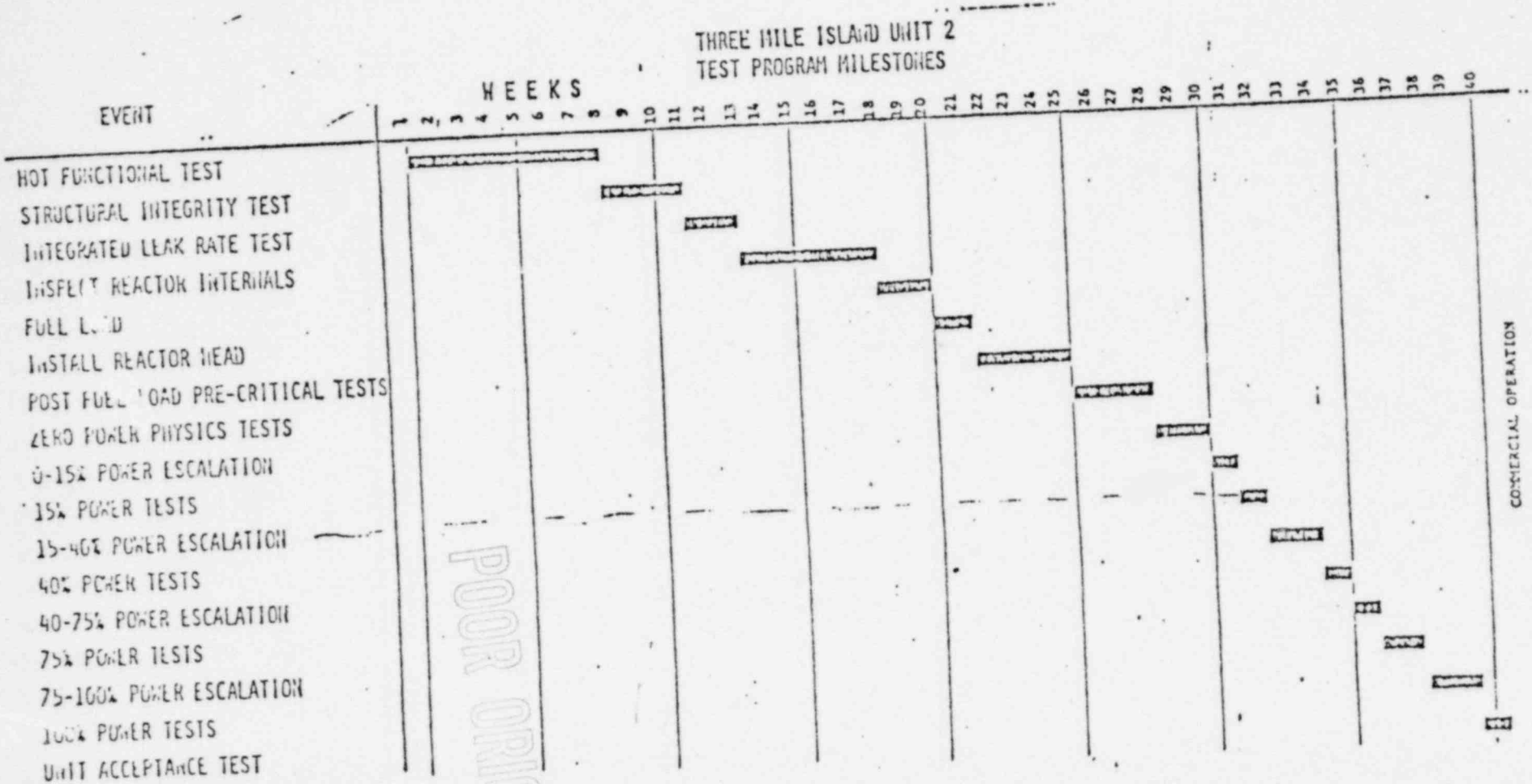
H. Dieckamp

lda
attachments

cc: Honorable Robert K. Bloom
Honorable H. Wilson Goode
Honorable Michael Johnson
Honorable Helen B. O'Bannon

Messrs. A. W. Johnson
R. L. Packard
M. Seidel
M. P. Widoff

THREE MILE ISLAND UNIT 2
TEST PROGRAM MILESTONES

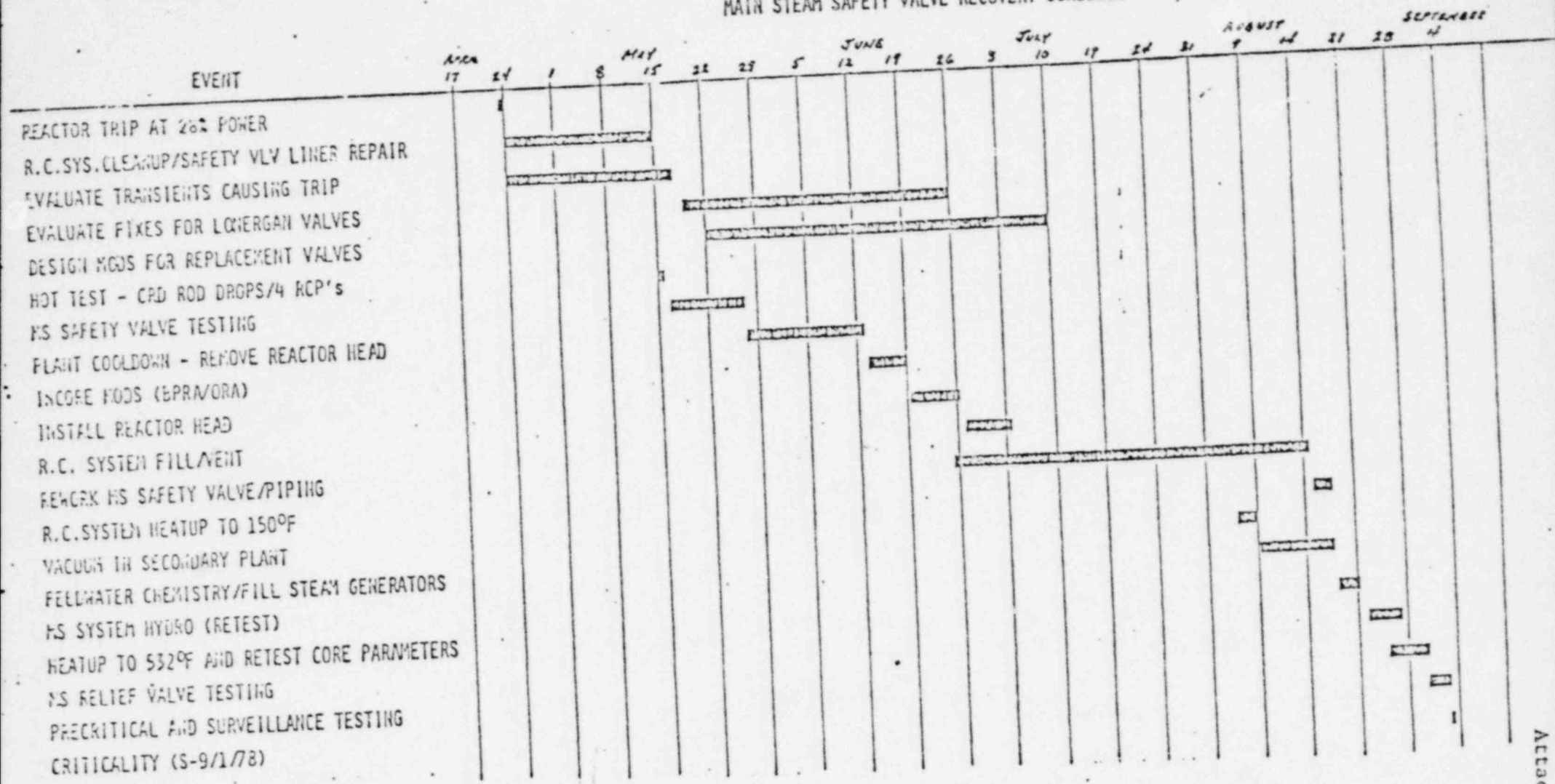


Attachment 1

PROBLEMS ENCOUNTERED DURING TEST PROGRAM

	CRITICAL PATH DELAY TIME
FUEL TRANSFER MECHANISMS	3 DAYS
STEAM GENERATOR INSTRUMENTATION PENETRATIONS	8 DAYS
REACTOR COOLANT PUMP MOTOR REVERSE ROTATION PROJECTION	3 DAYS
VITAL POWER INVERTER TRIP/SAFETY INJECTION	10 DAYS
ROD WORTH MEASUREMENTS	5 DAYS
BURNABLE POISON ROD ASSEMBLIES	NONE
SMALL BREAK LOCA ANALYSIS	NONE
MAIN STEAM SAFETY VALVE PERFORMANCE	EST. 4 MONTHS

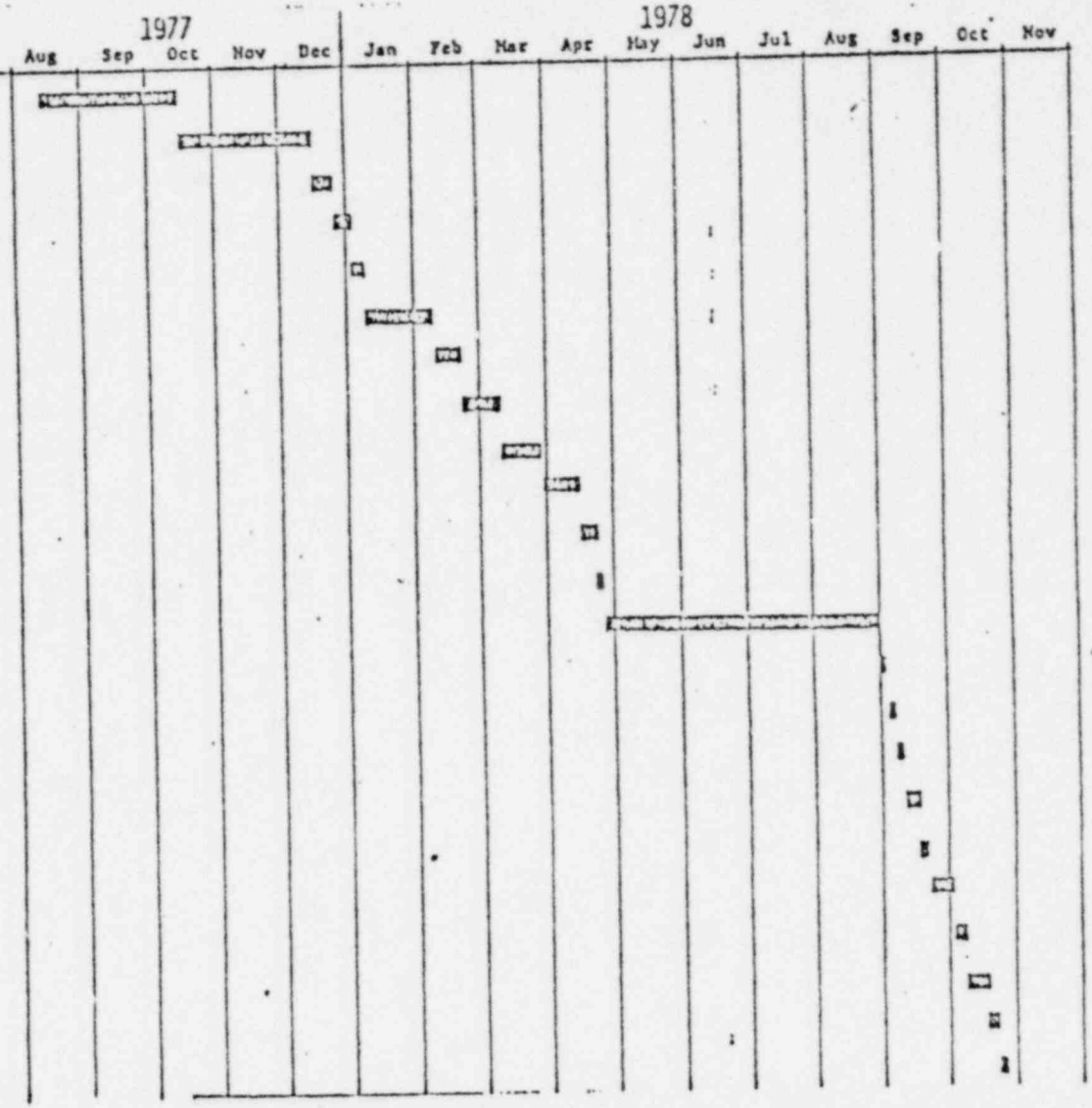
MAIN STEAM SAFETY VALVE RECOVERY SCHEDULE



Attachment 3

TEST PROGRAM AND MAIN STEAM SAFETY VALVE RECOVERY SCHEDULE

EVENT
1. HOT FUNCTIONAL TEST
2. RC PUMP REWORK/READINESS FOR STRUCTURAL INT.TEST
3. SAFETY FEATURES ACTUATION SYSTEM
4. STRUCTURAL INTEGRITY TEST
5. INTEGRATED LEAK RATE TEST
6. LICENSING PREREQUISITES
7. LOAD FUEL/INSTALL REACTOR HEAD
8. POST FUEL LOAD PRE-CRITICAL TEST (COLD)
9. POST FUEL LOAD PRE-CRITICAL TE[illegible] (HOT)
10. ZERO POWER PHYSICS/INITIAL CRITICALITY (3/28/78)
11. 0-15% POWER TEST/SYNC GRID ESCALATE TO 30%
12. 4/23/78 REACTOR TRIP AT 28%
13. MS SAFETY VALVE MOD/RETURN CRITICAL
14. ZERO POWER PHYSICS TESTING
15. 15% POWER PLATEAU
16. 15-[illegible]% POWER
17. 40% POWER PLATEAU
18. 40-75% POWER
19. 75% POWER PLATEAU
20. 75-100% POWER
21. 100% POWER PLATEAU
22. UNIT ACCEPTANCE TEST
23. COMMERCIAL OPERATION (S-10/20/78)



Attachment 4