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ENCLOSURES:

"Second Refueling Outage - Summary of Events of Interest"....

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CP&L

Carolina Power & Light Company

October 7, 1974

File: NG-3514 (R)

Serial: NG-74-1158

Mr. Edson G. Case, Acting Director
Directorate of Reactor Licensing
Office of Regulation
U. S. Atomic Energy Commission
Washington, D. C. 20545

Dear Mr. Case:

H. B. ROBINSON UNIT NO. 2
LICENSE DPR-23
SECOND REFUELING OUTAGE - SUMMARY OF EVENTS OF INTEREST

During the 1974 refueling outage of the H. B. Robinson Unit No. 2 Plant, several events transpired which were reported in accordance with Technical Specification and Bases or could be reported as items of general interest. A summary report covering these items has been prepared and is attached for your information. A list of the areas to be addressed is as follows:

1. Containment Integrated Leak Rate Test and Structural Integrity Test
2. Containment Liner Bulge Inspection
3. Inspection of Structural and Seismic Bolting of Class I Components
4. Pipe Hanger Inspections
5. Bolt Fragment Found in Steam Generator
6. Fuel Grid Damage
7. Retention of Control Rod in Upper Internals
8. Safety Injection Pump Test
9. Steam Dump Valve Modification
10. Fuel Inspection
11. Miscellaneous Refueling Items

We trust that this information will be of use to you.

DBW:mvp
Attachments

Yours very truly,

E. E. Utley
Vice-President
Bulk Power Supply

cc: Messrs. N. B. Bessac
W. B. Howell
J. B. McGirt
D. V. Menscer
N. C. Moseley
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H. B. ROBINSON UNIT NO. 2

SECOND REFUELING OUTAGE

SUMMARY OF EVENTS OF INTEREST

1. CONTAINMENT INTEGRATED LEAK RATE TEST (ILRT) AND STRUCTURAL INTEGRITY TEST (SIT)

The ILRT and SIT were performed in accordance with the H. B. Robinson Technical Specifications requirements in order to measure reactor containment building leakage at the peak calculated pressure of 42 psig, to establish a reference for subsequent periodic integrated leak rate tests at a reduced pressure of 21 psig, and to reconfirm the containment structural integrity. The test was performed as a joint venture with CP&L, Gilbert Associates, and Brewer Engineering Laboratories.

The reactor containment building was pressurized at a rate of 3 psi per hour. Building temperature was maintained at approximately 95°F. Pressure, temperature, recirculation unit motor current and compressor operation were monitored hourly.

The leak rate test was conducted at two pressure plateaus, 21 and 42 psig, with a plateau at 14 psig for visual internal inspection. During the reactor containment building internal inspection at 14 psig no leakage or other visible damage was found.

A minimum of four hours elapsed between stabilization of reactor containment building pressure and resultant data retrieval. During this period, and for the duration of the 24 hours leak rate and 12 hour supplemental test at each pressure level (21 and 42 psig), service water flow to the ventilating fans was varied to maintain average internal containment temperature within $\pm$ 0.2°F.

During each test the following occurred at half hour intervals:

a. Each of 6 dewcell temperatures was recorded and converted to water vapor pressure.

b. Each of 23 RTD temperatures was recorded and averaged.

c. Each of the 2 pressure gauge readings was recorded and averaged.

d. Pressure was corrected to obtain partial pressure of air and the weight of reactor building air was calculated. This weight was plotted along with the temperature.

The structural displacements and strains were measured at 0, 14, 21.1, 35, 42.2 and 0 psig by using direct current displacement transducers (DCDT's), strain gages, jig transits and scales, and invar tapes. The displacements and strains were checked for credibility and against the acceptance criteria at each pressure plateau before moving to the next pressure plateau. Visual inspections were made to check for gross or unusual deformations.

Results of the tests are as follows:

ILRT

Leakage Rate, Percent Per Day

L_t = max. allowable leak rate at 21 psig = 0.057

0.75 L_t = max. allowable measured leak rate at 21 psig = 0.042

L_{tm}, at 95.3°F (based on mass plot) = measured leakrate at 21 psig = 0.029

L_a = max. allowable leak rate at 42 psig = 0.080

0.75 L_a = allowable measured leakrate at 42 psig = 0.060

L_{am}, at 95.8°F (based on mass plot) = measured leakrate at 42 psig = 0.015

SIT

Crack patterns and widths, observed and measured during the structural integrity retest, compared well with data from the initial SIT. Crack widths were about the same. Generally the crack width range was from less than 0.005 inch to approximately 0.010 inch. Radial and vertical displacement data from the structural integrity retest showed general agreement with the initial SIT data and was within the limits of the acceptance criteria except for one radial measurement point, LC4. This point exceeded the acceptance criteria limits by 34 percent at 42 psig. However, if data for point LC4 is viewed with respect to all data for azimuth C, it appears to fit into the displacement curve. The same type of displacement curve was also observed at azimuth D.

Measured displacements and crack widths at design pressure indicated no significant difference from the measured values obtained during the structural integrity test of 1970 and satisfied the acceptance criteria. Thus, the continued structural integrity of the reactor containment building was demonstrated.

A detailed report enumerating methods of measurement, data accumulated, and evaluation of results shall be submitted as required by Technical Specification requirements. This brief summary only highlights the tests that were performed.

2. CONTAINMENT LINER BULGE INSPECTION

During this outage an effort was made to determine the integrity of the containment liner and to establish a program of surveillance using the base line data obtained during this inspection.

The inspection revealed bulges were present in the liner. One of these locations, covering an area of approximately 7 feet by 14 feet, was chosen for a detailed study. The insulation covering the liner was removed and it was found to be deformed toward the center of the containment approximately 2.5 inches from the theoretical curvature. There was a void area between the liner and the concrete as determined by the sound of tapping the area with a hammer.

On May 28, 1974, an Ebasco engineer, who was the original designer of the liner, examined the liner bulges. A detailed investigation was initiated, which is outlined below.

1. The liner was ultrasonically tested by Automation Industries to locate the studs securing it to the containment walls. The studs are on 16 inch centers which is more conservative than the original design which calls for 20 inch centers.

2. Further UT of the studs revealed indications at 4¼ inch to 4½ inch in all but three studs which had indications at 2 inches. Since the studs in this area are 4 inches long, it was concluded that all but three studs were unbroken.

3. Tapping with a hammer and using the hammer as an impact pendulum revealed that all the studs in the void area (even those with the 2 inch indication by UT) appeared to be solid.

4. Field tests led to the tentative conclusion that a pressure in excess of 75 psig would be required to push a stud surrounded by a deep void through the liner. Field tests also tentatively showed that the 42 psig pressure would cause a bump in the liner at a stud point surrounded by a deep void. Since there are no small bumps in the liner, it was concluded that the void was shallow and no danger existed of a liner rupture caused by a stud during a LOCA.

5. The Q.A. files refer to bulges in containment of the same magnitude and in the same general area. The files and people present during construction also mentioned void areas. These and other factors have led to the conclusion that the bulges are "as-built."

6. The engineering evaluation was performed by Ebasco. Their computer analysis led to the conclusion that the liner is safe for operation even if two adjacent studs are assumed broken. No corrective action is recommended.

7. The liner bulge has been instrumented with strain gages by Brewer Engineering Laboratories and was observed frequently during the containment heat-up and will be observed periodically thereafter. The purpose of the strain gages is to determine if the bulge is a passive as-built bulge as assumed; if not, to determine the growth rate and possible causes of the bulge.

It is concluded that the liner is safe for continued operation. Continued monitoring, both visually and with strain gages, during the next operating cycle will establish the active or passive nature of the bulge and the requirement for additional analyses.

3. INSPECTION OF STRUCTURAL AND SEISMIC SUPPORT BOLTING OF CLASS I COMPONENTS

In response to DRO Bulletin 74-3 a visual inspection was performed on all steam generator and reactor coolant pump support structure bolting.

There was no indication of loose or damaged bolting on any supports. However, one bolt was missing on "B" steam generator upper support. A stress analysis was performed by the design AE, Ebasco Services, Inc., and it was confirmed that the integrity of the support was not jeopardized with the bolt missing. Also some four washers were missing on the steam generator supports and four bolts were not fully seated due to an interference with adjacent fillet weld. An evaluation of these conditions show that they are not detrimental to the performance of the support system.

4. PIPE HANGER INSPECTION

The plant's inservice inspection schedule specified an inspection of pipe hangers to be completed during this refueling outage. This inspection was accomplished in compliance with the requirements of Section XI of the ASME Power Piping and Boiler Codes. No major discrepancies were noted as a result of the inspection. Various hangers were adjusted and hanger fasteners replaced or tightened. All existing hangers were determined to adequately support their respective piping sections. There are several deficiencies regarding the hangers that are yet to be repaired due to accessability and material problems. However, these areas do not jeopardize the safety of the system.

An inspection of hydraulic snubbers was also conducted during the outage. This inspection was in response to Regulatory Operations Bulletin 73-3 and 73-4. Several snubbers were lubricated at their pivot points and damping fluid was added as required to restore proper levels. No deficiencies were noted.

5. BOLT FRAGMENT FOUND IN STEAM GENERATOR

During eddy current inspection operations a bolt fragment was found in the inlet side of "A" steam generator channel head.

The bolt fragment consisted of a three-inch long piece of material, 5/8 inches in diameter. The head of the bolt had been severed from the bolt fragment by what appeared to be a saw cutting action. Approximately two inches of the fragment was threaded with eleven threads per inch (a standard coarse thread for 5/8 inch diameters) and the threads were badly abraided or missing in a number of areas and appeared to have been clamped in a vise. The fragment was activated to a radiation level of 2.5 roentgens indicating that it was exposed to the nuclear core prior to being deposited in the steam generator. (It would be expected that the radiation level would be approximately 200-300 MR if it had only been in the steam generator during this cycle; hence, the supposition that it had been in the core for some period of time.) The material was determined to be nonmagnetic and presumably is a variety of stainless steel.

Since the shank of the bolt fragment was the same diameter as the threaded portion and the fabrication techniques used for reactor internals bolts causes the shank to be less than the major diameter of the thread, the fragment does not appear to be from the reactor internals. Furthermore, a review of the possible bolts from the reactor coolant pumps was made, and samples of the most probable bolts were obtained from Westinghouse EMD. It has been confirmed by comparison with the fragment removed from the steam generator, that the fragment is of a different type than those used in the pumps.

With respect to the absence of battering of the bolt fragment and the steam generator tube sheet, it is believed that the fragment dropped into the manway well when it was first carried into the channel head by the coolant flow. This is a low flow area, and it is felt that the coolant velocity in this immediate area was not sufficient to move the bolt fragment any further.

Therefore, the bolt does not appear to have originated in the reactor coolant system and is considered a foreign object. No apparent damage was sustained by the system due to the presence of this object.

6. FUEL GRID CLIP DAMAGE

As previously reported, two sections of fuel clips were found in the "C" steam generator channel head during the November, 1973, eddy current inspection. Efforts during the refueling outage were directed to location of the source of these fragments and their accountability.

Via television inspection it was verified that the fragments were dislodged from the sixth grid, faces 1 and 2 of fuel assembly C-08. Refer to Figures 5 and 6 for the location of the displaced straps. One section of the strap on face 1 was not accounted for. The section was approximately ¼ inch by 3/4 inch. Adjacent fuel assemblies, B-06, C-43 and B-52, were inspected, and no damage to these assemblies was detected.

In regard to the portion of the strap that was not located, it was assumed that it may still be in the reactor core. An analysis of this situation was then made based on this assumption. It was concluded that debris of this type could be postulated to cause a control rod drive or scram problem if a piece becomes lodged above the core in a control rod guide tube of the reactor internals or in a guide thimble in the fuel assembly. However, the probability of this occurring is low because the time the piece would remain in the upper core region is expected to be small. Coolant flow would most likely sweep the piece into a loop and it would ultimately be trapped elsewhere: steam generator, lower plenum of the reactor vessel or core.

There is no apparent safety problem with a control rod, since any interference would probably be limited to one rod at a time. This is not an unreviewed concern, since it is assumed that one control rod does not function when calculating shutdown margin. If, however, two control rods were rendered inoperable, a highly unlikely occurrence, steps in accordance with Technical Specification 3.10.5 would have to be followed.

With respect to partial flow blockage caused by a missing grid fragment, which could occur if a piece becomes trapped in the bottom nozzle or between fuel rods, and grids, it is not considered that this would be a safety concern with respect to DNB.

In summary, debris of this size would not be expected to result in a safety problem in the fuel. Although precautions are taken to insure a debris-free system prior to operation, there have been cases where pieces of debris of this size are known to have been in the coolant during operation and no safety-related problems occurred. Furthermore, pieces which might still be unaccounted for have already been in the primary system for one complete cycle and may have already been removed from the core if entrapped in the two regions of discharged fuel.

7. RETENTION OF RCC IN UPPER INTERNALS

In conjunction with refueling operations the reactor vessel head was removed on May 22, 1974, and the cavity flooded early on the morning of the 24th. Shortly after noon on the 24th, the upper internals package was removed and stored in its rack in the north end of the reactor vessel cavity. At 0315 on May 25, 1974, the first fuel element was removed from the core. Fuel movement continued smoothly until 1412 on May 26, at which time, it was discovered that the RCC assembly was missing from Fuel Element B-4 which was removed from core position N-7. An investigation was immediately begun, and through visual inspection it was determined that the RCC rodlets were protruding about 36 inches sideways from beneath the upper internals package stored on the west side of the R.V. Cavity.

An evaluation was made at this time to insure adequate shutdown margin was available with this RCC out of the core. A boron concentration of 2154 ppm was measured which insured a shutdown margin greater than that required by Technical Specifications.

An attempt was made on Thursday, May 30, 1974, to unlatch the drive shaft from the RCCA but was unsuccessful due to the force being applied to the RCCA spider by the bent rodlets which were lying on the cavity floor. On Friday, May 31, 1974, the upper internals package was lifted, relieving the pressure on the spider, permitting the RCCA drive shaft to be unlatched from the spider hub. The drive shaft was removed from the guide tube and on June 1 and 2 borescope inspections were performed on the guide tube by Westinghouse personnel with no abnormalities observed. The lower portion of the upper internals package was inspected by use of an underwater T.V., with all areas observed showing no damage from the event.

On June 2 a visual inspection was performed on the drive shaft. Slight scarring was observed on the latching pins, therefore, this shaft was replaced with a spare.

A new RCCA was received from the Westinghouse fabrication shop in Columbia, South Carolina, and was installed in the fuel element to be inserted in core position N-7.

Core loading was completed and the upper internals packaged installed in the Reactor Vessel on June 9, 1974. All drive shafts were latched and tested as per Refueling Instruction, F.T. 9.7. As an additional precaution, the RCCA in position N-7 was withdrawn and inserted three additional times with weight and drag being observed and recorded to insure proper operation.

The reactor Vessel Head was replaced on June 10, 1974, and preparation made to continue refueling maintenance activities.

This event has been reviewed and no malfunction of components or equipment has been observed. Although procedures were followed and check off sheets completed, it must be assumed that through operation error, the step in the unlatching process was performed out of sequence which allowed the shaft to be relatched while disconnecting the tool from the shaft.

To prevent recurrence of this type event, the procedure will be revised with the final step requiring a "tool only" weight observation to insure unlatching has indeed taken place.

8. SAFETY INJECTION PUMP TEST

As a part of the outage maintenance all three safety injection pumps were disassembled and inspected. Pump shaft seals were replaced at that time. A crack in the "B" pump eight stage impeller was also weld repaired and "B" pump impeller shaft was replaced. Previous repairs in November of 1973 consisted of replacement of the "A" pump impeller shaft.

Following the maintenance effort a performance test was run. The purpose of the test was to repeat the test performed in April, 1973, that is; (1) measure the pump shutoff head (at miniflow), (2) measure the system flow delivery under the condition of essentially zero reactor backpressure, (3) demonstrate that flow is delivered down each individual cold leg injection branch line, (4) demonstrate the effect of the pump miniflow on the system delivery and also to compare the performance of the safey injection pumps before and after they had been refurbished.

The test results indicated that the total developed head for each of the three safety injection pumps, while running on miniflow, was above the minimum acceptance value of 1425 psig and therefore in accordance with the Technical Specification requirements. For all three pumps, the measured TDH was found to be somewhat higher than had been measured during a similar test in 1973 and before the pumps were refurbished. Refer to attached Figures 1, 2, and 3.

The measurement of the flow rate through each of the cold leg branch lines verified the pipe resistance calculations that the branch to loop 3 had the lowest resistance and compared favorably with the results of the 1973 test. This branch line giving the highest flow was assumed to spill during an accident and the safety analyses were performed utilizing the sum of the flows from the two remaining branch lines for emergency core cooling.

The required flow to the reactor cooling system through the three cold leg injection lines from two operating pumps was demonstrated by the test for each pair of pumps, i.e. the system flow was greater than the acceptance value of 839 gpm.

The results plotted on Figure 4 indicate that the system performance for the minimum safeguards case used in the safety analyses again was verified by the test. The system delivery calculated from the test data was slightly greater at the higher reactor backpressure which represents a small improvement in the speed at which the boron injection tank contents would be swept into the reactor during a steam break accident.

The small effect of the pump miniflow was demonstrated during the acceptance runs with two pumps operating since the system delivery changed only slightly when the miniflow was terminated.

Again as had been determined in the 1973 test, the pump performance over the entire operating range was found to be less than had been expected during the plant design. Some small improvement of pump performance seems to be indicated since the 1973 test.

9. MAIN STEAM DUMP AND CONDENSER DUMP VALVES

The plant's condenser dump valves (PRV-1324-A-1, 1324-A-2, 1324-B-1, and 1324-B-2) had experienced numerous problems regarding performance and reliability. Specifically they would not modulate in their low range of operation, experienced excessive oscillation and experienced seat leakage. Attempts had previously been made to correct these problems with the addition of valve dampers, by-pass lines, and control system changes. However, these modifications had not proved successful. A study was made to completely overhaul the system, and it was decided to replace the four 12-inch valves with five 8-inch valves to provide better seating abilities and made control of the valves easier. This modification was accomplished during the outage. The four valves were replaced and a fifth valve was added on the west end of the condenser. The addition of the fifth valve required the installation of some 100 feet of additional piping. Piping modifications were also required to install the new valves. This consisted of the installation of piping reducers to accomodate the 8-inch valves, slight rerouting of the piping, and the installation of additional hangers. Also new controller systems were installed on all five valves.

During the 1973 refueling outage the atmospheric dump valves (PRV-1325-A, 1325-B, 1325-C, and 1325-D) were removed from the system and returned to the vendor for overhaul. The plant was operated during cycle II without the valves. The valves were repaired and reinstalled during the 1974 outage. Quick-change valve seats were installed on the valves, and they were converted to a flanged connection configuration. The valve pneumatic controls were also modified by conversion to nitrogen supply and the addition of nitrogen accumulators. These changes improve the valve serviceability, seat tightness, and control reliability.

Following the above modifications the subject valves were test stroked at 0 psig and 1000 psig. Test signals were also simulated in the electrical control circuitry and the valve response checked. No large load reduction has been imposed on the new system.

10. FUEL INSPECTION

Using underwater TV inspection equipment, all Region 4 and certain Region 2 and 3 fuel assemblies were inspected to confirm that fuel to be used during Cycle 3 was capable of additional exposure. Inspection at that time and subsequent review of the videotapes of the inspection by CP&L personnel indicated that Region 4 had no anomalies or failures, with only a minimal amount of rod bowing.

Several assemblies, including three from Region 4, were observed to have bent top nozzle springs (See table 2). Since no assembly had more than one bent spring and three springs are sufficient to hold an assembly against the lower core plate even under the postulated conditions of pump overspeed, the bent springs do not present any problem for Cycle 3. A study is in progress, however, to identify and eliminate the cause of the bent springs. It was concluded that all Region 4 assemblies are fully capable of operating through the end of Cycle 3.

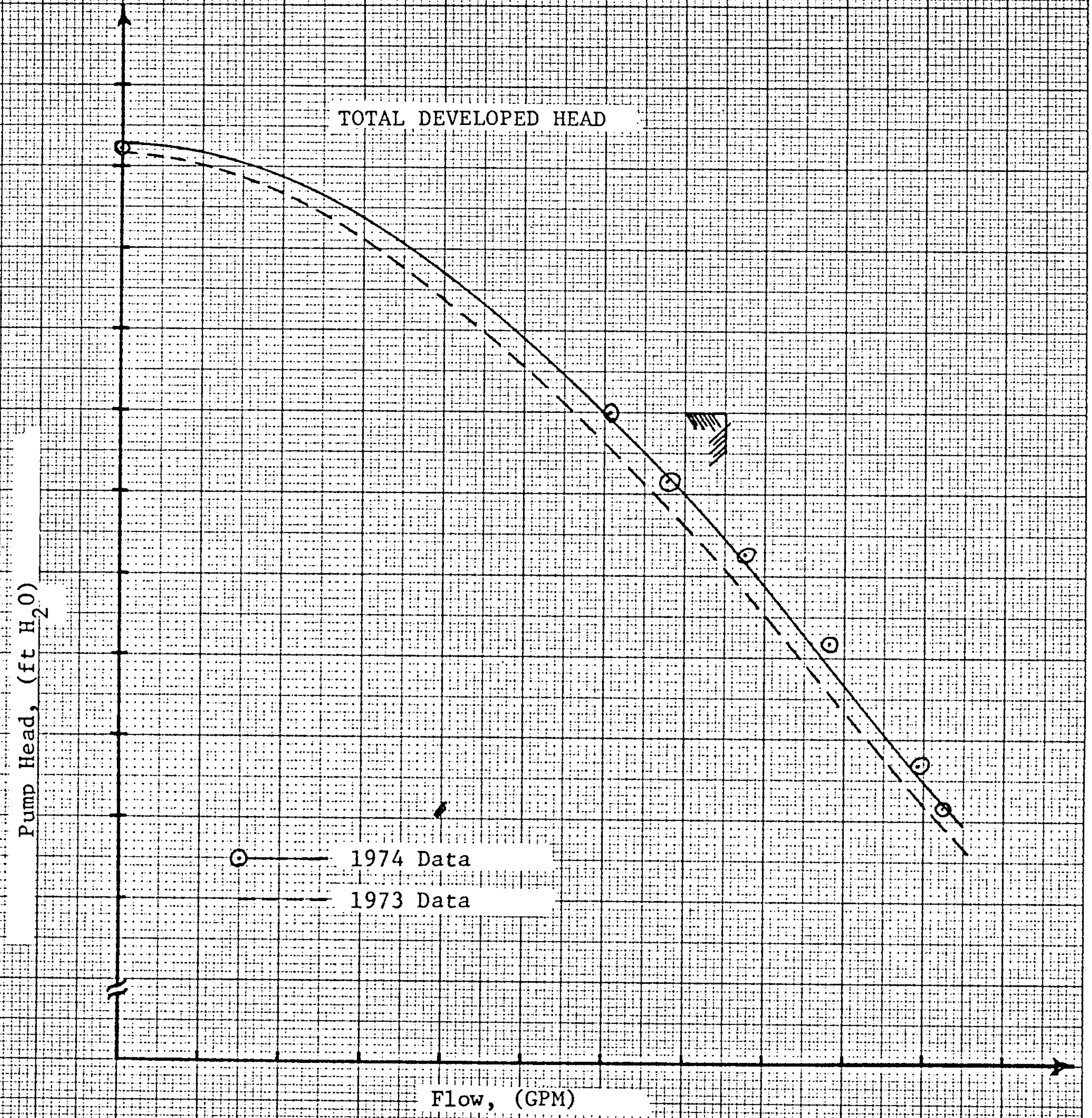
Seven RCC's were inspected using the CP&L underwater TV equipment. With the appropriate fuel assembly located in the RCC change fixture, the RCC hub and spider were checked for cracks, dents, wear and braze joint condition. The RCC was then withdrawn and the rodlets checked for wear, unusual crud patterns, or other anomalies. The seven RCC's inspected (See table 1) were selected based on core location and operating history.

All RCC's inspected were in good condition. All braze joints were intact and exhibited no evidence of cracks or other anomalies. No unusual wear patterns were noted on RCC rodlets.

11. MISCELLANEOUS REFUELING ITEMS

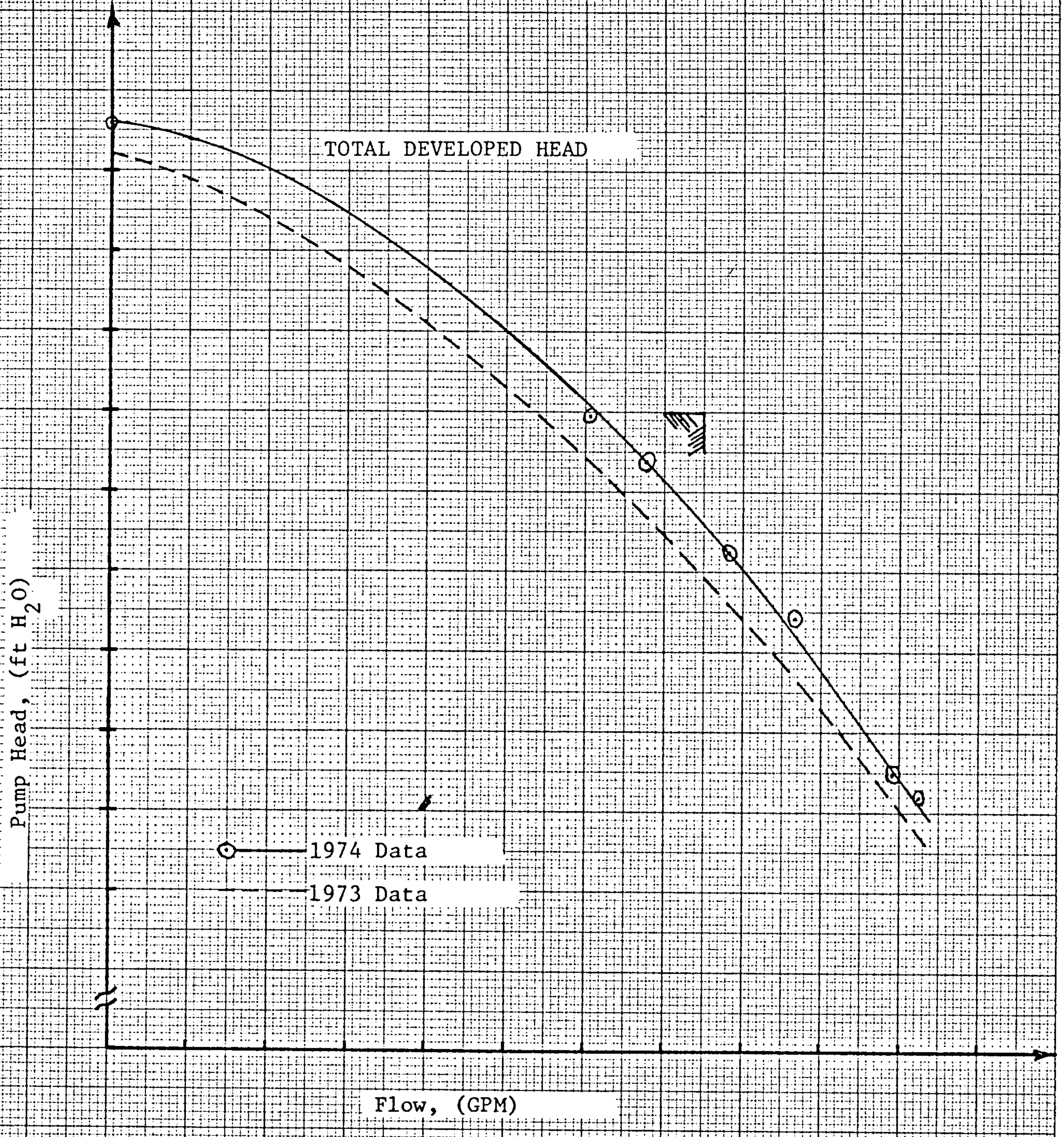
Periodic tests performed to verify setpoints of pressurizer safety valves and main steam safety valves revealed various valves that required adjustment. One of the pressurizer valves was found to open at 2550 psig, whereas the required setpoint is 2485 $\pm$ 25 psig. The valve was lapped and adjusted to the required setting. Eleven of the twelve main steam safety valves setpoints were out of the required range. The maximum deviation was valve 1A which was tested at 1190 psig, compared to its required range of 1085 psig $\pm$ 1%. All valves were adjusted to the proper setpoints.

Outage plans called for replacement of motor control center (MCC-5 and MCC-6) overcurrent trip devices. However, the breakers which were available were not compatible with the control centers. Therefore, this preventive maintenance item was deferred until the next refueling. Setpoints of the existing breakers were verified. The dashpots on the undervoltage trip coils were also inspected and were in good condition with no evidence of cracking.



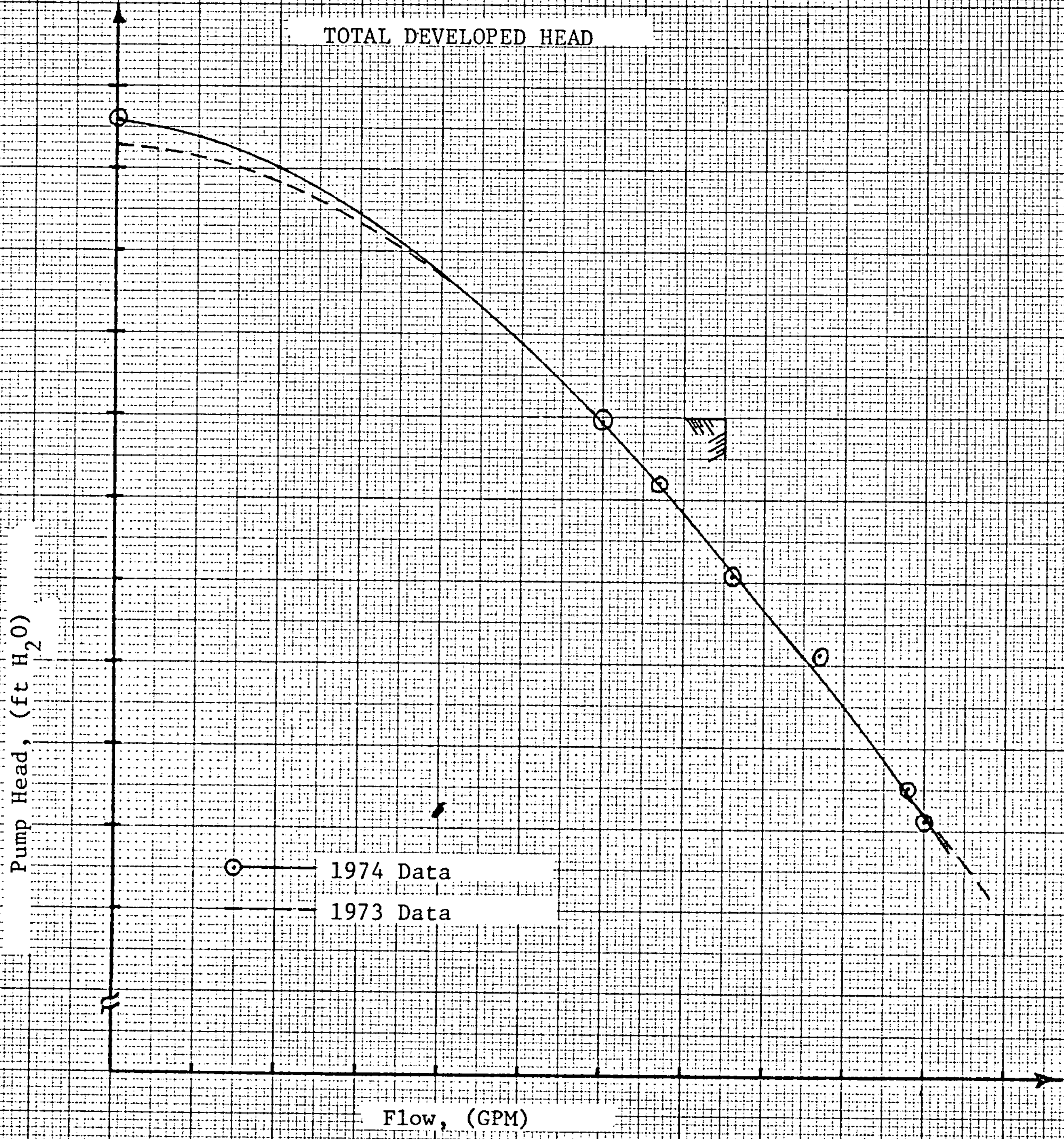
CP&L SAFETY INJECTION PUMP A
Design Point = 375 GPM, 2500 ft)

Figure 1



CP&L SAFETY INJECTION PUMP B
Design Point = (375 GPM, 2500 ft)

Figure 2



CP&L SAFETY INJECTION PUMP C

Design Point = (375 GPM, 2500 ft)

Figure 3

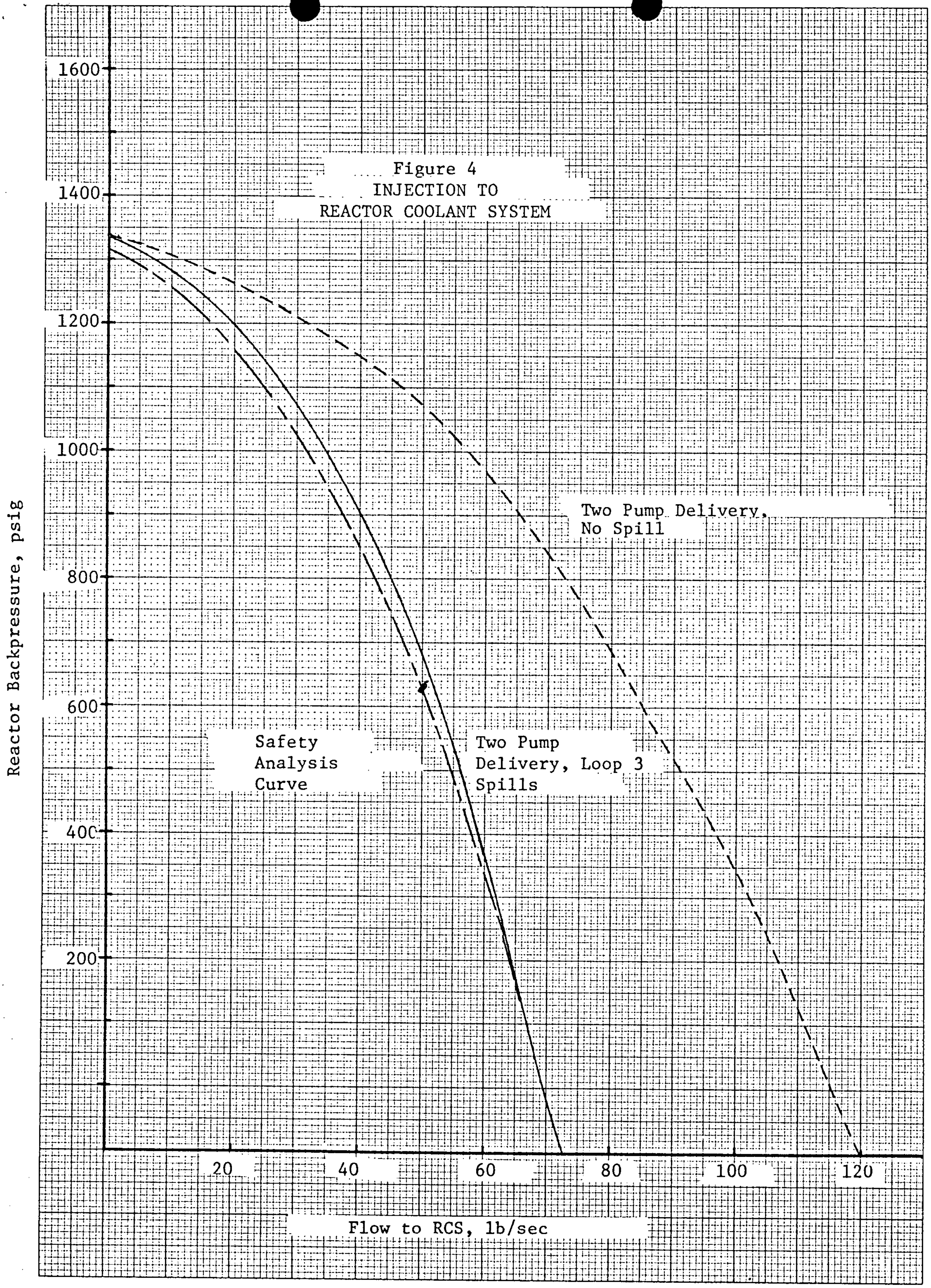


Figure 4
INJECTION TO
REACTOR COOLANT SYSTEM
Reactor Backpressure, psig
1600
1400
1200
1000
800
600
400
200
Two Pump Delivery,
No Spill
Safety
Analysis
Curve
Two Pump
Delivery, Loop 3
Spills
20
40
60
80
100
120
Flow to RCS, lb/sec

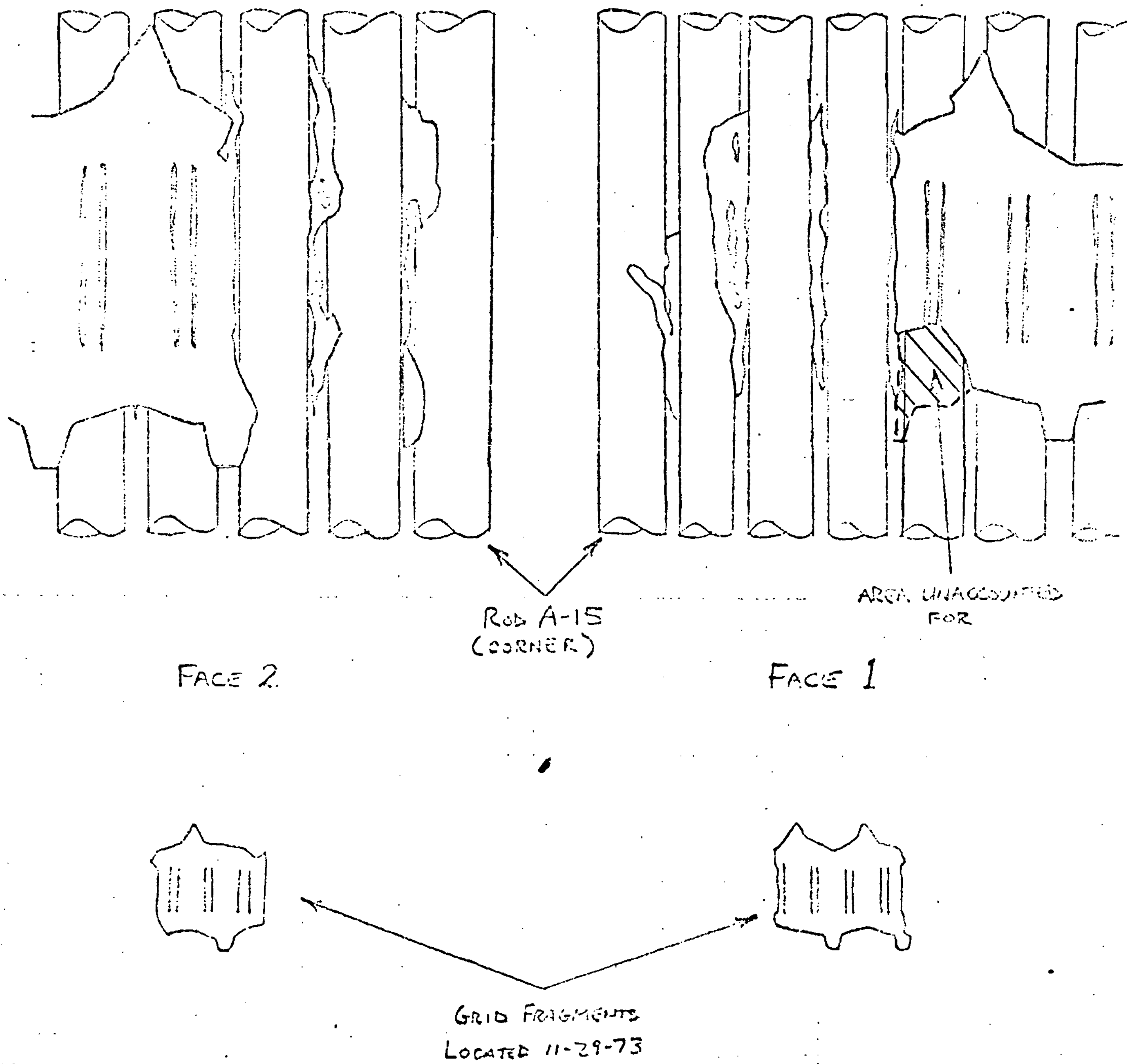


FIGURE 5

FUEL ASSY. C-08

GRID #6

6-1-74

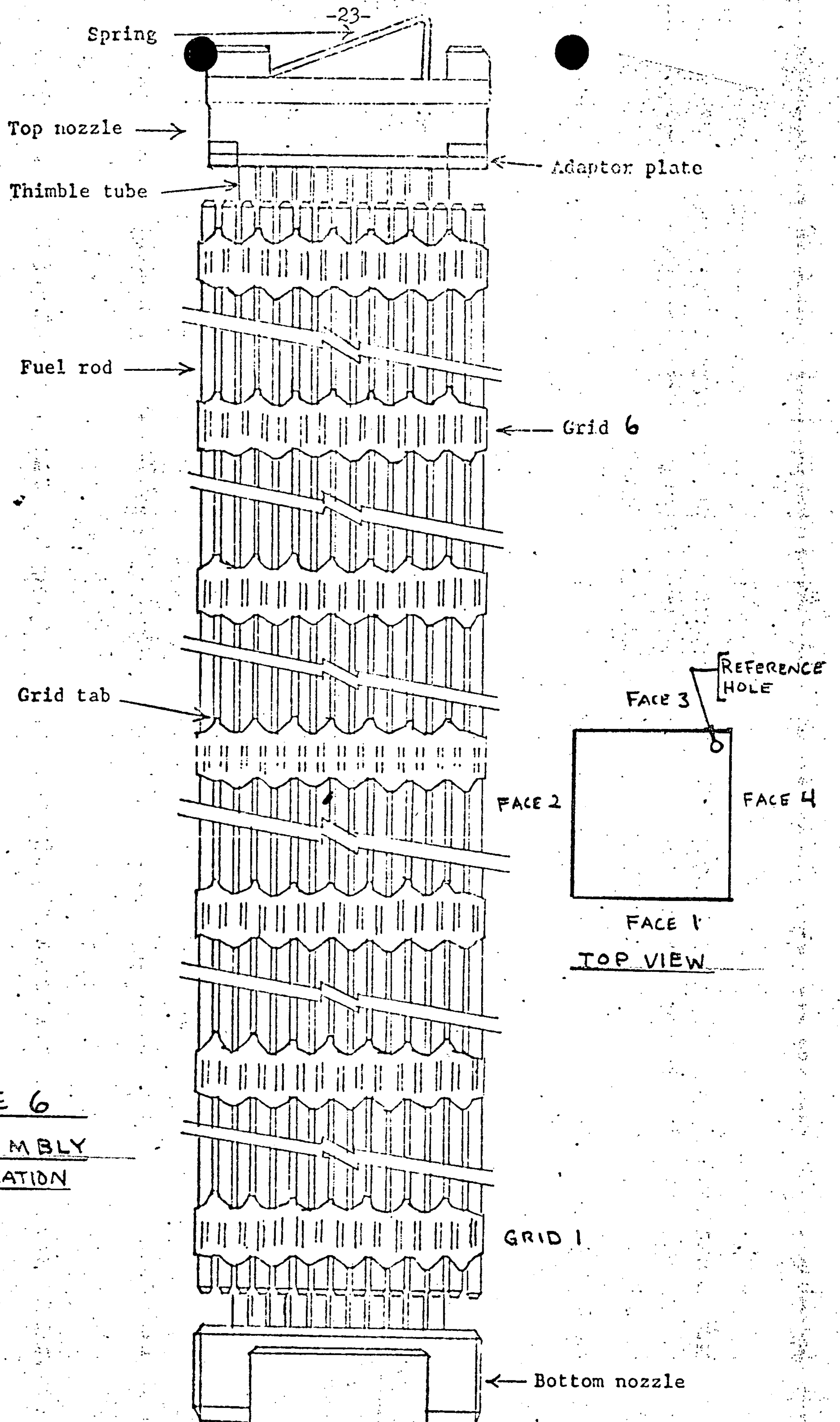


FIGURE 6

FUEL ASSEMBLY IDENTIFICATION

Table 1

RCC's Inspected During Cycle 2-3 Refueling Outage

R16
R41
R35
R47
R22
R5
R6

Table 2

Region 4 Assemblies With Bent Top Nozzle Hold-Down Springs

	Comment
D-19	Spring on 180° face depressed 3/4 to 1 inch
D-53	Spring on 180° face depressed 1 inch
D-44	Spring on 90° face depressed ½ inch