CONSUMERS POWER COMPANY
Big Rock Point Nuclear Plant

SPECIAL REPORT

Big Rock Point Plant Operations

1. Reactor Water Chemistry-Fuel Crud Deposition
2. Cycle 10 Fuel Performance
3. Cycle 11 Start-Up Report

Prepared by: Consumers Power Company
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8101100409

I. REACTOR WATER CHEMISTRY-FUEL CRUD DEPOSITION

A. INTRODUCTION

Enough experience and data has been accumulated over the years to convince us that we now understand the full dimensions of the Big Rock Point reactor water chemistry-fuel crud deposition problem. Most important of all, we feel we have implemented satisfactory solutions. However, only future fuel performance can confirm this judgment.

The sources of the impurities making up the fuel crud composition have been identified and have been eliminated (refer to Table 1). In addition, the peculiar crud deposition pattern on the E-fuel has been remedied.

B. BACKGROUND

A review of the archives files on the Big Rock Point Plant show the reactor water chemistry and crud deposition problems to be the most persistent of the problems experienced at the plant over the years. The basic difficulty in defining and solving these problems has been related to our inability to get representative water samples for analysis from the primary and feed-water system and a good measure of the total amount of the crud deposition on the fuel. As a result of the recently completed extensive water chemistry program and fuel profilometry study, enough quantitative data are now available to draw some firm conclusions.

The first fuel inspections at Big Rock Point revealed crud on the fuel. Chemical analysis showed the crud to consist mainly of zinc, nickel, iron and copper - clearly constituents of the feed-water heater tube material. In March 1968, the feed-water heater tube material was changed from copper-nickel to stainless steel. Subsequent feed-water testing confirmed that this change effectively eliminated the main source of copper, zinc and nickel in the feedwater.

During the short operations cycle (Cycle 5) following the change-over of the feed-water heaters, the water chemistry-crud deposition appeared to be well under control, and Consumers Power was confident that a significant corner had been turned even though Cycle 5 was of short duration (four months) and with a peak reactor power of only 60 MWe.

Cycle 6 (July 1968 to May 1969) started off at 75 MWe reactor power but because of high off-gas the load was reduced partway through the cycle to 60 MWe. It was also noted during this cycle that the reactor water pH varied considerably. This indicated that the primary system was seeking a new chemical equilibrium because of the changeover of copper-nickel feed-water tubes to stainless steel. At the end of Cycle 6, crud deposition measurements clearly showed that the situation was worse than ever. Additionally, the crud composition had changed to predominantly copper from the earlier zinc, nickel, copper composition.

An extensive cooperative water chemistry study program with General Electric Company was instituted for Cycle 7 (May 1969 to February 1970). This study showed that the feedwater had been cleaned up. There was some evidence that small amounts of impurities were injected from the demineralizers to the primary system after "fluffing" of the beds, shut-downs and/or regeneration of the resins but this could not account for the levels of copper seen in the reactor water. The conclusion was that the source of copper had to be in the primary system. The only possible sources were "hideout" on the surfaces of the primary system including the steam drum and the copper alloy tubes in the clean-up system heat exchangers (both regenerative and nonregenerative).

The reactor water chemistry appeared to stabilize during Cycle 7. Crud deposition measurements at the end of Cycle 7 showed an improvement in the deposition rates. This improvement continued through Cycle 8 (March 1970 to February 1971). The total amount of crud deposited on fuel rods dropped significantly from Cycle 6 to Cycle 7, then again from Cycle 7 to Cycle 8. However, the peak crudded area of the fuel (the lower quarter) still exhibited the same peak rod crud deposition rate.

During Cycle 7, a series of tests was run on water entering and leaving the clean-up system heat exchangers. These tests revealed that the water was picking up copper from the copper alloy tubes in the clean-up system heat exchangers. Profilometry studies on the crud deposition patterns in fuel assemblies showed that crud was preferentially being deposited on the lower quarter of the fuel assemblies on the outer row of fuel rods facing the channel walls. Knowing the crud thickness and this pattern made it possible by using mass balance techniques to conclude that most of the copper in the crud deposited on the fuel could be attributed to the copper picked up in the clean-up system heat exchangers.

At the end of Cycle 7, the internal surfaces of the steam drum were inspected. The inspection proved the steam drum to be relatively clean with no large visible accumulations of crud. Considering the purity of the feedwater, it was concluded that the clean-up system heat exchangers constituted the major source of copper for the crud deposition.

Also, during Cycle 8, it was determined that there was a significant bypass flow (approximately 13 gpm) between the tube and shell sides of the clean-up system regenerative heat exchangers. Previously, during Cycle 7, a small leak rate had been calculated (approximately 2 gpm). This increase in bypass flow further confirmed that the copper-nickel tube clean-up system regenerative heat exchangers were a source of copper in the reactor water.

C. PROBLEM

Only in mid-1971, after all the above experience and data had been accumulated, could the full dimensions of the reactor water chemistry-crud deposition problem be appreciated.

First, the source had to be identified, but it changed with time. Initially, it was the feed-water heaters; then it was "hideout" or just an inventory of feed-water heater tube material in the primary system. These sources masked two other lower level sources that probably contributed impurities from "day one." These other sources were the clean-up system heat exchangers and the impurities washed off the condensate demineralizers by "fluffing," after regeneration or after a shutdown.

Secondly, some explanations for the crud deposition pattern on the fuel were to be provided to help explain the cause of the fuel failures. Initially, on the B- and C-fuel, there was so much crud available it deposited fairly uniformly over the fuel. (Crud depositions were not as limiting on B- and C-type fuels as they are on E- and F-type fuels because of greater heat transfer areas associated with the B- and C-fuels.) Then, as the primary system cleaned up a bit after the feed-water heater changeover to stainless, the crud began to deposit preferentially in lower positions yet higher power regions of the fuel rods (exterior fuel bundle rods). The flow tests also explained in part the preferential accumulation of crud on the lower quarter of the outer fuel rods.

D. CORRECTIVE ACTION

The solution to the reactor water chemistry-crud deposition problem consisted basically of eliminating the source of the impurities that made up the crud. The following have been done:

1. The feed-water heater tube bundles were changed to stainless steel in March 1968.

2. Operating practices with the condensate demineralizers were reviewed and modified. After May 1970, the resins were no longer regenerated. This effectively eliminated the "spikes" noted in the water chemistry study program.

3. The clean-up system heat exchangers were replaced in April 1972.

In addition to the above, the original fuel channel-orifice hardware on 69 of the 84 fuel support-tube-and-channel assemblies has been replaced during the refueling outages in 1972 and 1973. The modified support-tube-and-channel assemblies improve the flow characteristics in the lower quadrant of the fuel bundle.

E. RESULTS

Fuel performance during Cycle 10 (May 1972 - February 1973) improved markedly. More heat was produced with smaller off-gas releases than in the several previous cycles. Continued improvement in fuel performance is expected as copper "hideout" sources are depleted and existing fuel that has been exposed to previous water chemistry conditions is replaced by new fuel.

II. CYCLE 10 FUEL PERFORMANCE (MAY 1972 - FEBRUARY 1973)

A. INTRODUCTION

Cycle 10 fuel examination was conducted during the March 1973 refueling outage. The primary purposes of the irradiated fuel inspection were to characterize Cycle 10 fuel performance, obtain information concerning crud deposition and to enable the respective fuel supplier to collect data to verify several of their design models such as fuel rod growth, clad creepdown and pellet stack shortening.

Consumers Power Company, General Electric Company, Exxon Nuclear Company and Battelle Northwest Laboratories participated in the fuel examination. The inspection activities were performed by two independent teams. The first consisted of General Electric personnel and the second was comprised of individuals from Exxon Nuclear and Battelle Northwest. Consumers Power Company's participation in the examination consisted of performing dry sipping tests on all assemblies scheduled to be returned to the core; scheduling and coordinating the examination activities; assisting the inspection teams in performing specific tests; and, analysis of selected portions of the data.

B. WORK SCOPE

The inspections performed consisted of visual examinations of exterior surfaces of selected fuel assemblies and fuel rods; profilometry; gamma scanning; eddy-current testing; and, fuel rod length measurements. The specific work scopes are detailed below:

1. General Electric

a. Visual Examination

General Electric Company and Consumers Power Company performed a visual examination of ten fuel assemblies containing individual rod failures to determine the failure mechanisms. The fuel assemblies included in the examination were four (4) Type F, three (3) EEI mixed-oxide and three (3) modified EG assemblies. The exposure range for these assemblies was 6,300-15,700 MWd/T. A visual examination of twelve (12) removable EEI mixed-oxide rods was also performed.

b. Profilometry Analysis

Twelve (12) fuel rods and fourteen (14) nonfueled rods (cobalt target) from F-type assemblies were removed and profiled to obtain information concerning crud thickness, clad creepdown and clad ovality.

c. Length Determination

Length measurements were performed on the fourteen (14) nonfueled rods to determine Zircaloy clad growth.

d. Mixed-Oxide Fuel Examination

The following examinations were performed by General Electric on mixed-oxide fuel:

(1) Visual examination of the three EEI mixed-oxide fuel assemblies and twelve EEI mixed-oxide removable rods.

(2) Sipping tests were performed on eight of the EEI removabl rods to determine cladding integrity. In an attempt to locate the failed rod(s) in assembly EP-02, all eight gasoline-uranium oxide rods were removed and the bundle was resipped. The resulting sip signal showed the assembly to be failed, thus indicating a failed Pu rod.

(3) Two mixed-oxide rods from assembly EP-02 were profiled to obtain data on crud thickness.

2. Exxon Nuclear

Exxon's fuel inspection focused on three (3) assemblies (D-70, -71, and -72) and a total of twenty (20) individual fuel rods withdrawn from these assemblies. In addition to the examinations described in Paragraphs a and b under "General Electric," Exxon performed the following tests on the twenty (20) withdrawn rods:

a. The fuel rod lengths were measured to determine the amount of rod growth.

b. Eddy-current and ultrasonic testing were performed to locate and define cladding defects.

c. The rods were gamma-scanned to detect changes in the pellet stack length and to verify that there were no significant gaps in the pellet stack.

d. Of the 20 fuel rods inspected, two were mixed-oxide rods. These two rods underwent the same examinations as the uranium rods.

3. Results of the Cycle 10 Fuel Examination

a. Fuel Failure

The sipping results revealed that 23 of 84 fuel assemblies contained failed rods. The end-of-life exposures of 17 of the failed assemblies exceeded 10,000 MWd/T. Visual examination of the failed assemblies revealed the probable cause of the failures to be accelerated corrosion. Evidence of internal hydriding was not observed. Table 1 provides a detailed description of the Cycle 10 core composition and identifies the assemblies that failed.

b. Mixed-Oxide Fuel Performance

The three (3) mixed-oxide fuel assemblies (EP-01, -02 and -03) also contained failed rods. Their bundle exposures were approximately 15,400 MWd/T, 15,700 MWd/T and 15,000 MWd/T, respectively.

Assembly EP-01 was visually inspected in an attempt to determine the cause of failure. Only one rod exhibited slight crud spalling (located in the middle of the lower tier). All the other sides of the assembly appeared normal. Visual examination of the peripheral rods did not reveal the cause of the failure.

Assembly EP-02 likewise was visually inspected. Slight crud spalling was observed on one exterior rod. All other sides of the assembly appeared normal. A further attempt was made to determine the location of the failed rod(s). All eight (8) gadolinia-uranium oxide rods were removed and the assembly was resipped. The resulting sip signal showed the assembly to be failed, indicating a Pu rod failure. In addition to the above inspections, two exterior rods were profiled to determine crud thickness. The crud thickness noted is equivalent to that found on other 3-cycle fuel assemblies.

Fuel assembly EP-03 was reconstituted using EP-01 as a source of acceptable fuel rods. The fuel rods removed from EP-03 exhibited crud spalling and white zirconium oxide. These observations indicate that the rod failures were due to accelerated corrosion of the cladding from the outer diameter surface. This type of failure mechanism has been noted on uranium oxide fueled rods and is presently the predominant cause of fuel failures at Big Rock Point.

Twelve (12) EEI mixed-oxide rods from various uranium fueled assemblies were also examined. Two of the rods were withheld from future operation since clad corrosion attack was observed. These two rods were not sipped. The exposure for these two rods was approximately 17,000 MWd/T each. The remaining ten rods were found to be sound. General Electric selected four of the ten rods for shipment to UNC for post-irradiation examination. The remaining six rods were returned to the core for further irradiation.

Two mixed-oxide rods from assembly D-72 were examined by Exxon Nuclear. The scope of the inspection is given in Section II, Paragraph B. Both of these rods were sound.

c. Crud Deposition

A comparison between profilometry data for Cycles 9 and 10 revealed that the peak diametrical crud buildup has decreased from approximately 0.006 inch to 0.0045 inch. An increase in loose flaky crud was observed for Cycle 10, indicating that the crud morphology is being altered to a more permeable and less tenacious nature.

d. Effect of Redesigned Support-Tube-and-Channel Assemblies

In 1972, forty (40) redesigned support-tube-and-channel assemblies were inserted in the core. Their effect on fuel performance cannot be evaluated at this time because of insufficient information. Higher exposures are required on fuel assemblies that have been restricted to operation in the new channels before a judgment can be made.

4. Conclusion

Since it has been shown that the presence of deposited crud on fuel rods is contributing to excessive cladding corrosion, decreases in deposited crud, changes in crud morphology and improvements in feed-water impurity conditions are significant. These effects should provide for improved fuel performance during Cycle 11. Analysis of Cycle 10 fuel performance indicates the absence of failures due to internal hydriding of zirconium.

Analysis of examination data indicates that mixed-oxide rods fail from a mechanism identical to that causing uranium fuel rod failures. When these failures have occurred, no plutonium has been detected outside of the fuel rods.

With known defective mixed-oxide fuel in the Big Rock Point reactor, we have been unable to identify any plutonium in the primary coolant, liquid or gaseous effluent. Plutonium would be most readily detected through alpha activity. Measurement of gross alpha radioactivity is conducted on all effluents. If one made the very unrealistic assumption that every alpha particle detected came from plutonium-239, the maximum liquid effluent concentrations during the past several years would be:

Maximum Alpha Activity - 2 cpm/ml

This is equivalent to approximately 2.7×10^{-6} µCi/ml in the waste tank or approximately 5×10^{-10} µCi/ml in the discharge canal. The permissible public drinking water limit for plutonium-239 is 5×10^{-5} µCi/ml.

Selected rods from the modified EG, high burnup Reload C, the corner rod experiment carrier assemblies, and four (4) EEI mixed-oxide rods were shipped to Vallecitos for possible postirradiation destructive examination. General Electric intends to obtain up to seven (7) additional mixed-oxide rods for possible postirradiation examination.

Future examination (Cycle 11) will center on the 11 x 11 fuel. This fuel has greater thermal margin (lower average heat flux) than our present 9 x 9 fuel and will significantly decrease problems due to accelerated cladding corrosion.

III. CYCLE 11 START-UP REPORT

A. INTRODUCTION

Big Rock Point Cycle 11 was designed to operate between 200 and 220 MWT for approximately one year. The core loading includes eight (8) General Electric Reload EG assemblies; sixty-five (65) General Electric Reload F assemblies; two (2) Exxon Nuclear J1 assemblies; two (2) Exxon Nuclear J2 assemblies; one (1) EEI-General Electric EP assembly; four (4) Nuclear Fuel Services Demonstration Assemblies (NFS-DA); and two (2) Exxon Nuclear prototype Reload G assemblies. The prototype Reload G and the NFS-DA are new fuel types being irradiated for the first time at Big Rock Point. Both are 11 x 11 array designs while the remainder of the core consists of 9 x 9 array fuel types that have been used previously. The J2, EEI, NFS-DA and G assemblies all incorporate mixed-oxide fuel in their designs.

B. REACTOR START-UP

Upon completion of core reconstitution, core shutdown margin verification was successfully performed as required by Technical Specifications Section 5.2.2(b). Each step of the verification was comprised of the complete withdrawal of one strong peripheral control rod plus an adjacent control rod withdrawn six notches. Computer calculations showed the extra six notches to be worth in excess of 0.4% $\Delta k/k$. Additional shutdown margin tests were conducted by entirely withdrawing two adjacent

TABLE 1

End of Cycle 10 Core Composition

Bundle Type	Exp MWd/T	No of Cycles	Status	Bundle Type	Exp MWd/T	No of Cycles	Status	Bundle Type	Exp MWd/T	No of Cycles	Status
F-01	6,654	2	F	F-29	3,477	1		D-72	5,646	1	
F-02	10,418	2	F	F-30	4,848	1		D-73	5,598	1	
F-03	10,708	2		F-31	4,997	1		EP-01	15,416	3	F
F-04	10,018	2		F-32	4,397	1		EP-02	15,712	3	F
F-05	8,986	2		F-33	1,951	1		EP-03	15,026	3	F
F-06	2,251	1		F-34	2,766	1		D-61	12,333	3	F
F-07	10,043	2		F-35	5,073	1		D-62	12,418	3	F
F-08	9,770	2		F-36	4,115	1		D-63	11,837	3	F
F-09	9,606	2		F-37	5,267	1		E-55	7,767	2	
F-10	9,964	2		F-38	5,436	1		E-61	16,390	4	F
F-11	10,148	2		F-39	2,794	1		E-62	16,662	4	F
F-12	6,289	2	F	F-40	2,839	1		E-65	11,579	3	
F-13	9,430	2		F-41	5,448	1		E-66	11,635	3	F
F-14	8,757	2		F-42	5,244	1		E-67	11,956	3	F
F-15	10,833	2		F-43	4,042	1		E-68	11,927	3	
F-16	9,946	2		F-44	5,014	1		E-70	7,346	2	F
F-17	9,694	2		F-45	2,711	1		E-71	11,667	3	F
F-18	8,748	2	F	F-46	1,980	1		E-72	12,750	3	
F-19	6,547	2	F	F-47	4,470	1		E-74	14,360	3	
F-20	9,578	2		F-48	5,041	1		E-75	12,669	3	F
F-21	9,222	2		F-49	4,832	1		E-78	12,507	3	
F-22	10,346	2		F-50	2,254	1		E-79	13,769	3	
F-23	9,636	2		F-51	4,532	1		E-80	13,003	3	
F-24	6,289	2	F	F-52	2,263	1		E-81	14,067	3	F
F-25	9,032	2		F-53	1,990	1		E-82	12,807	3	
F-26	3,420	1		F-54	1,974	1		E-83	12,669	3	F
F-27	2,217	1		D-70	10,210	2		E-84	13,966	3	F
F-28	4,468	1		D-71	10,675	2		E-85	12,388	3	F

strong peripheral control rods plus withdrawal of six notches of a neighboring control rod. As predicted by computer calculations, the core remained subcritical during each step of the shutdown margin test. Final calculations indicated a total shutdown margin of 6% Δk/k with the most valuable control rod (3% Δk/k) withdrawn from the core.

The first Cycle 11 beginning-of-cycle (BOC) cold critical control rod pattern differed from the computer predication by one notch (~0.06¢ Δk/k). Figures 1 and 2 are diagrams of the predicted and actual Cycle 11 BOC critical control rod patterns.

After completion of the first critical approach, the moderator temperature coefficient test was conducted (Technical Specifications Section 5.2.4). Results of the test indicate a maximum addition of 12 cents from ambient (∿70°F) to 137°F, well within the Technical Specifications limit of one dollar. Figures 11 and 12 are plots of ρ vs temperature and the temperature coefficient vs temperature (ρ/T).

Fluxwires were irradiated upon reaching equilibrium conditions at a power level of 216 MWT. Figures 3 through 10 are comparisons of the actual vs computer-predicted axial flux distribution for the eight (8) in-core monitor locations. These figures indicate a very good match between the predicted and measured axial flux distributions.

In summary, the physics start-up was uneventful. All predicted and measured values were in good agreement and well within Technical Specifications.

Figure 1

Predicted BOC 11 Cold Critical
Control Rod Pattern

	A	B	C	D	E	F
1		8	10	8	10	
2	10	0	0	0	0	8
3	8	0	0	0	0	10
4	10	0	0	0	0	8
5	8	0	0	0	0	10
6		10	8	10	8	

BRP CONTROL ROD POSITIONS

Figure 2

Actual BOC 11 Cold Critical
Control Rod Pattern

	A	B	C	D	E	F
1		8	10	8	10	
2	10	0	0	0	0	7
3	8	0	0	0	0	10
4	10	0	0	0	0	8
5	8	0	0	0	0	10
6		10	8	10	8	

BRP CONTROL ROD POSITIONS

Super-critical On A Period of Approximately 60 Seconds

FIGURE

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

FLUXWIRES ○

COMPUTER PREDICTIONS △

FLUXWIRE DATA
4-26-73

LOCATION NO. ____

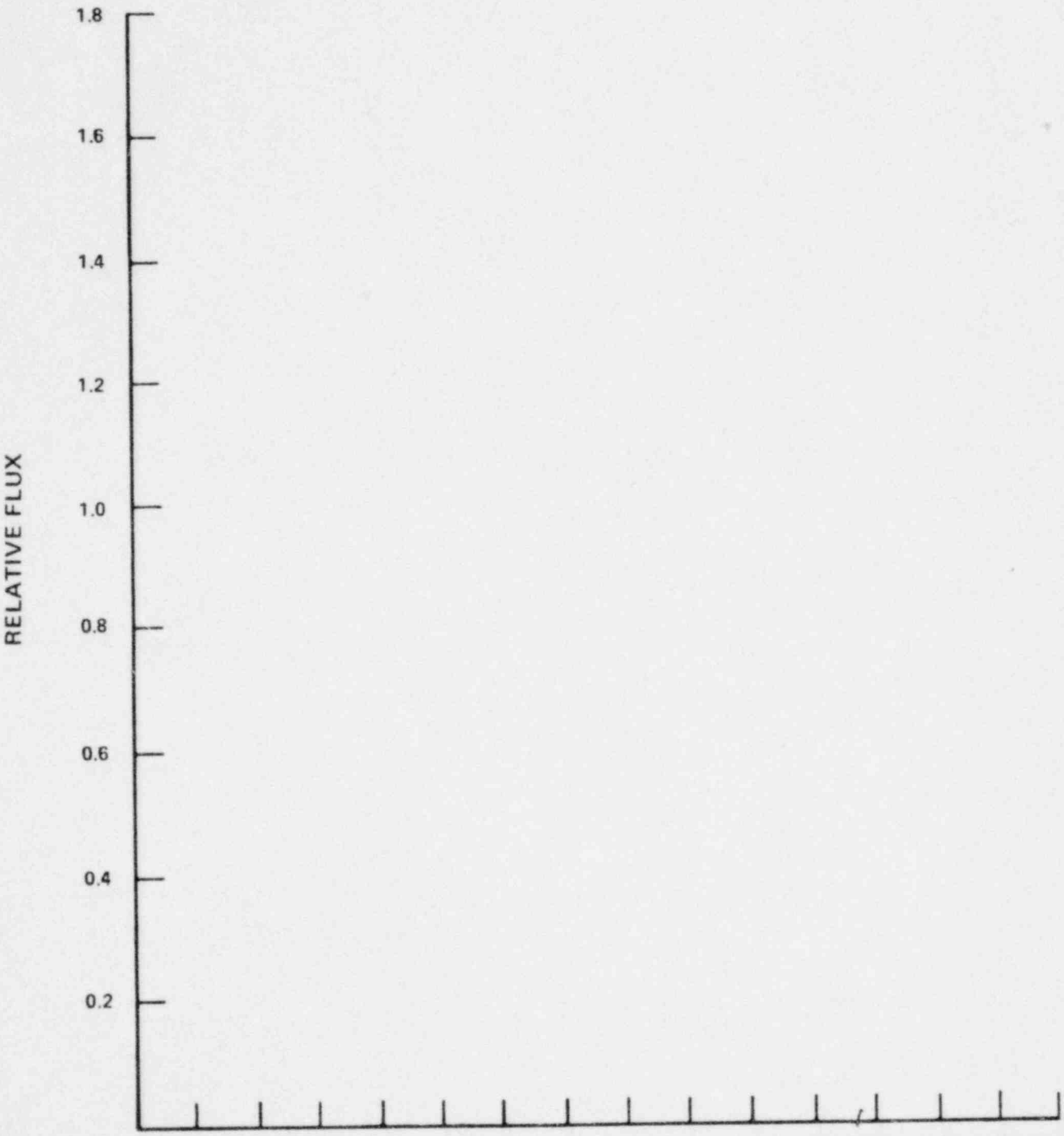
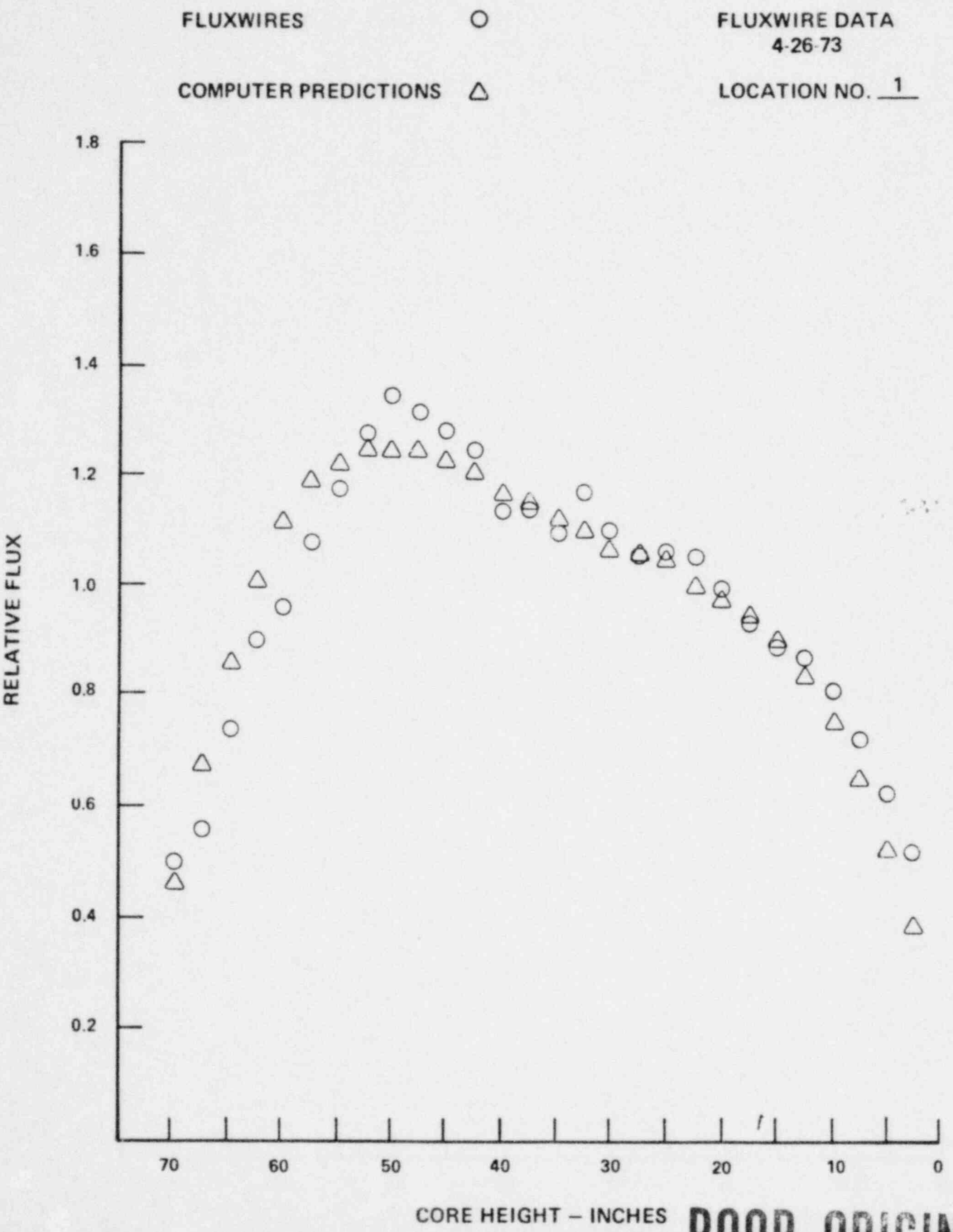


FIGURE 3

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

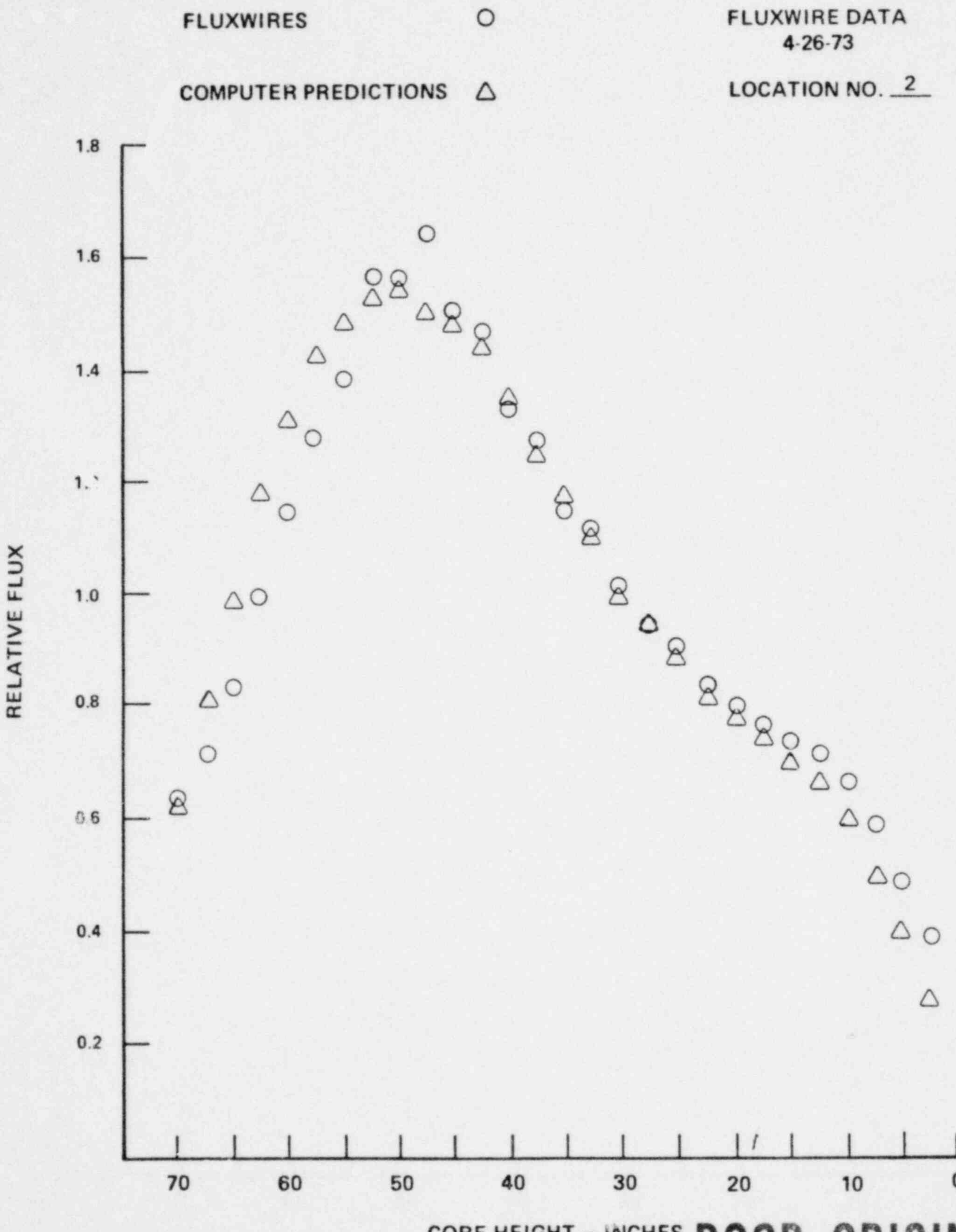


POOR ORIGINAL

FIGURE 4

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

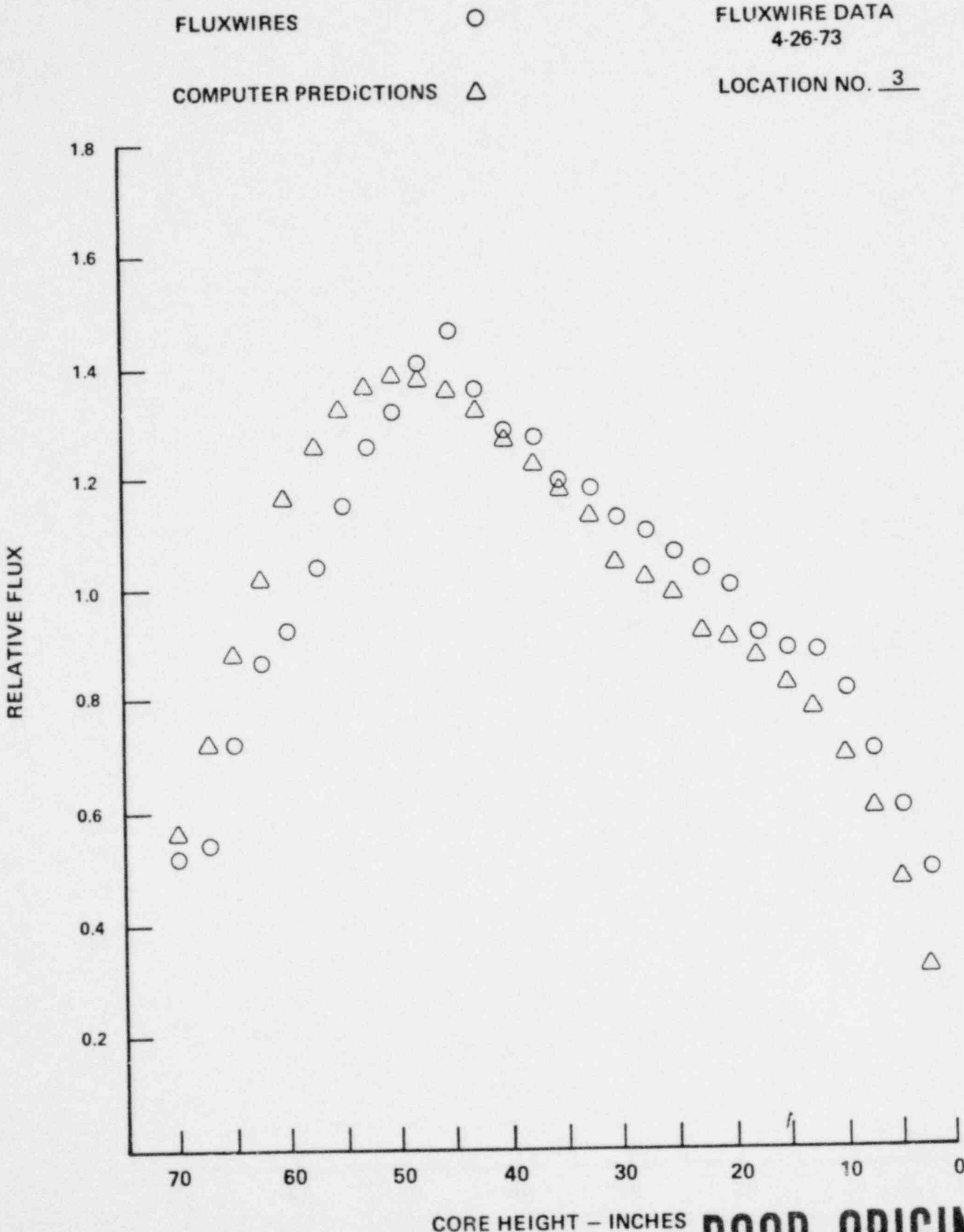


POOR ORIGINAL

FIGURE 5

NORMALIZED AXIAL PROFILE

BIG ROCK POINT



POOR ORIGINAL

FIGURE 6

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

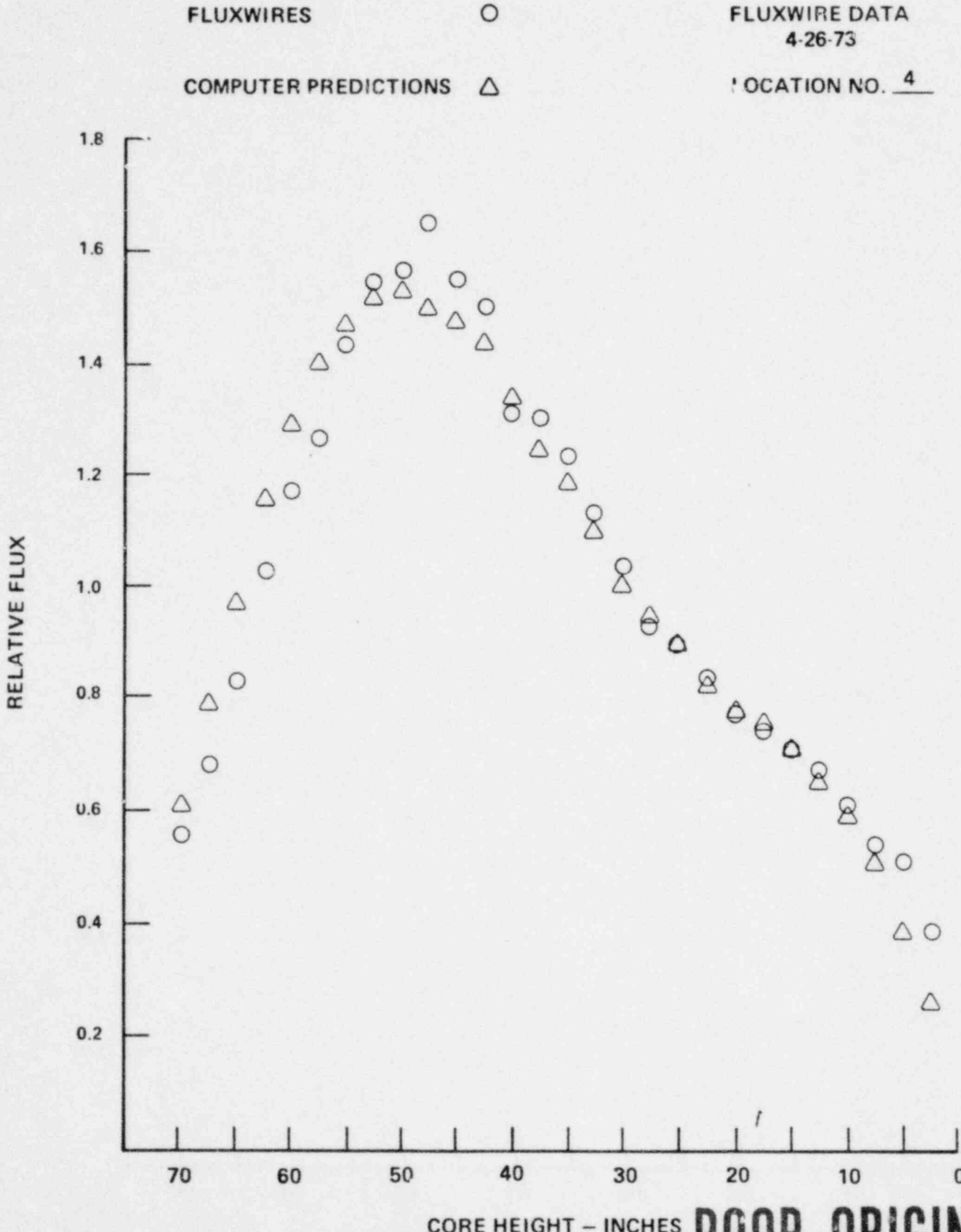
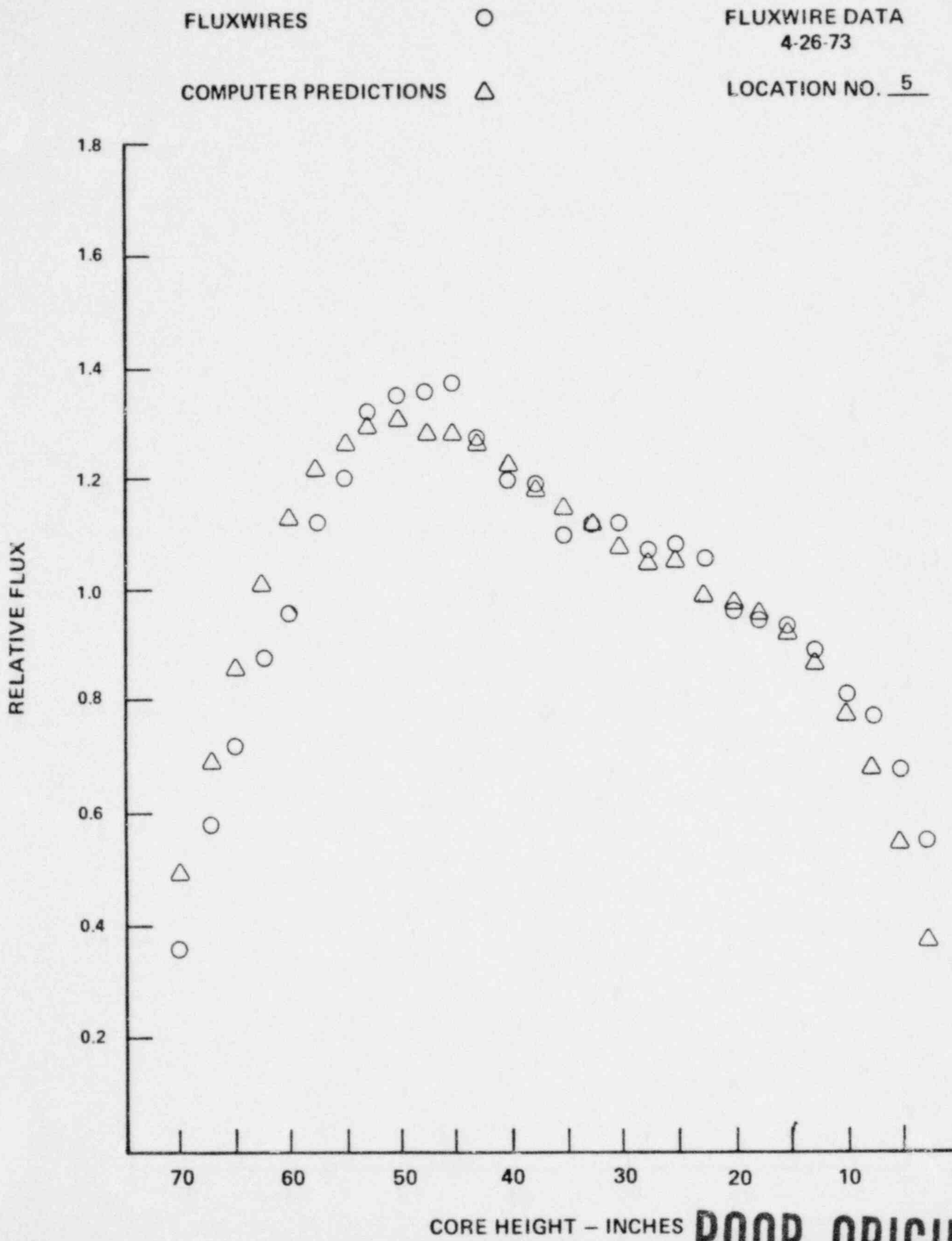


FIGURE 7

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

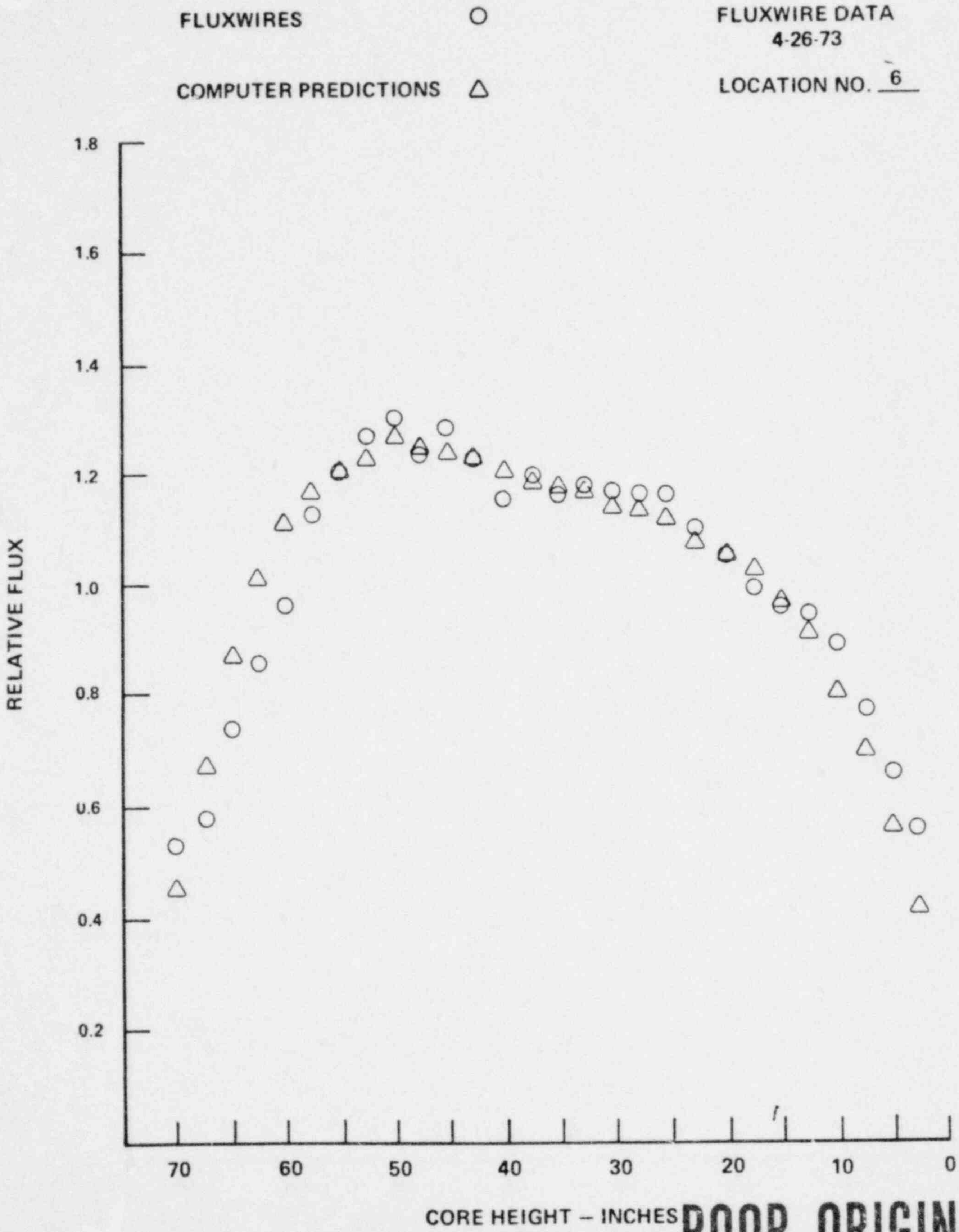


POOR ORIGINAL

FIGURE 8

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

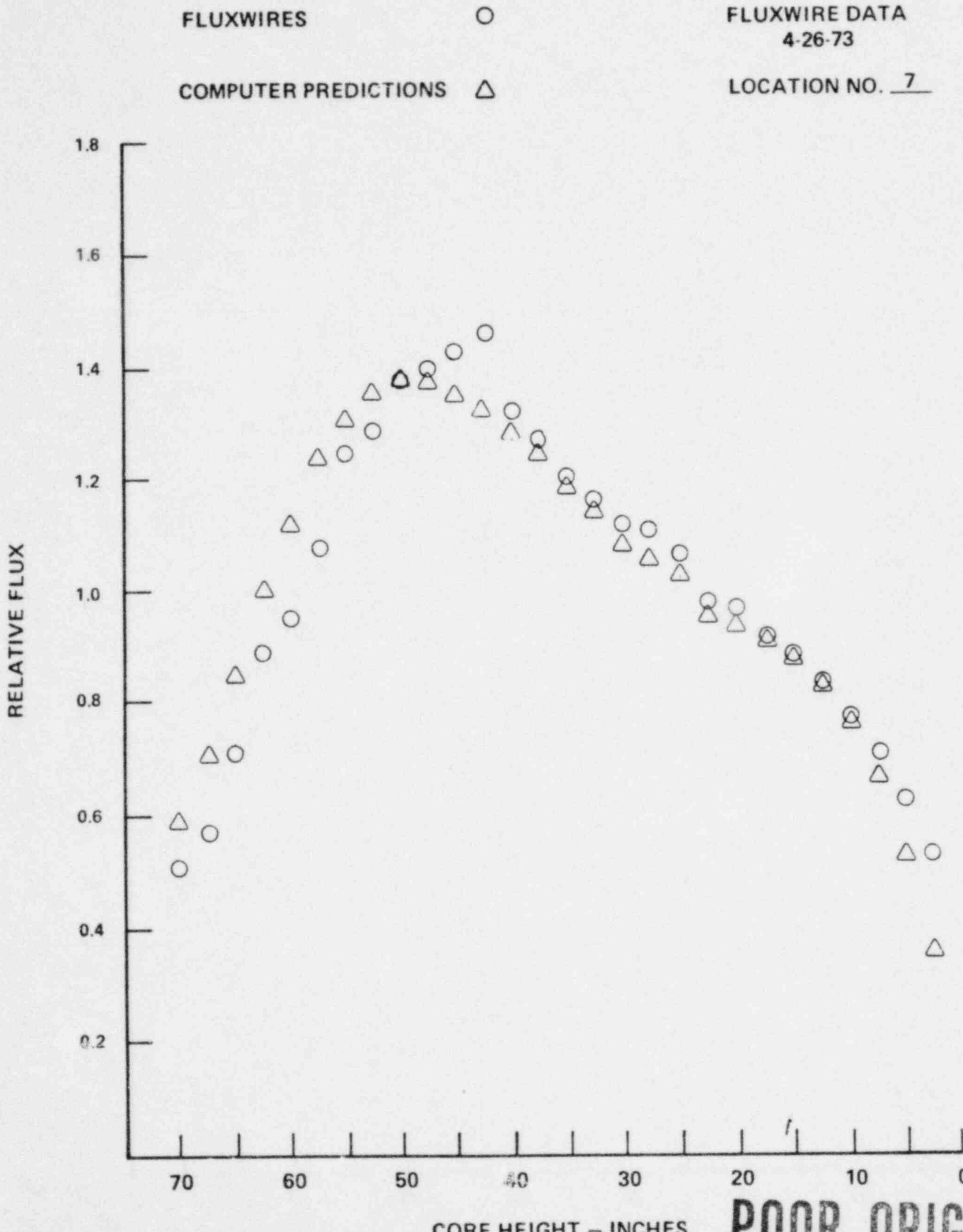


POOR ORIGINAL

FIGURE 9

NORMALIZED AXIAL PROFILE

BIG ROCK POINT

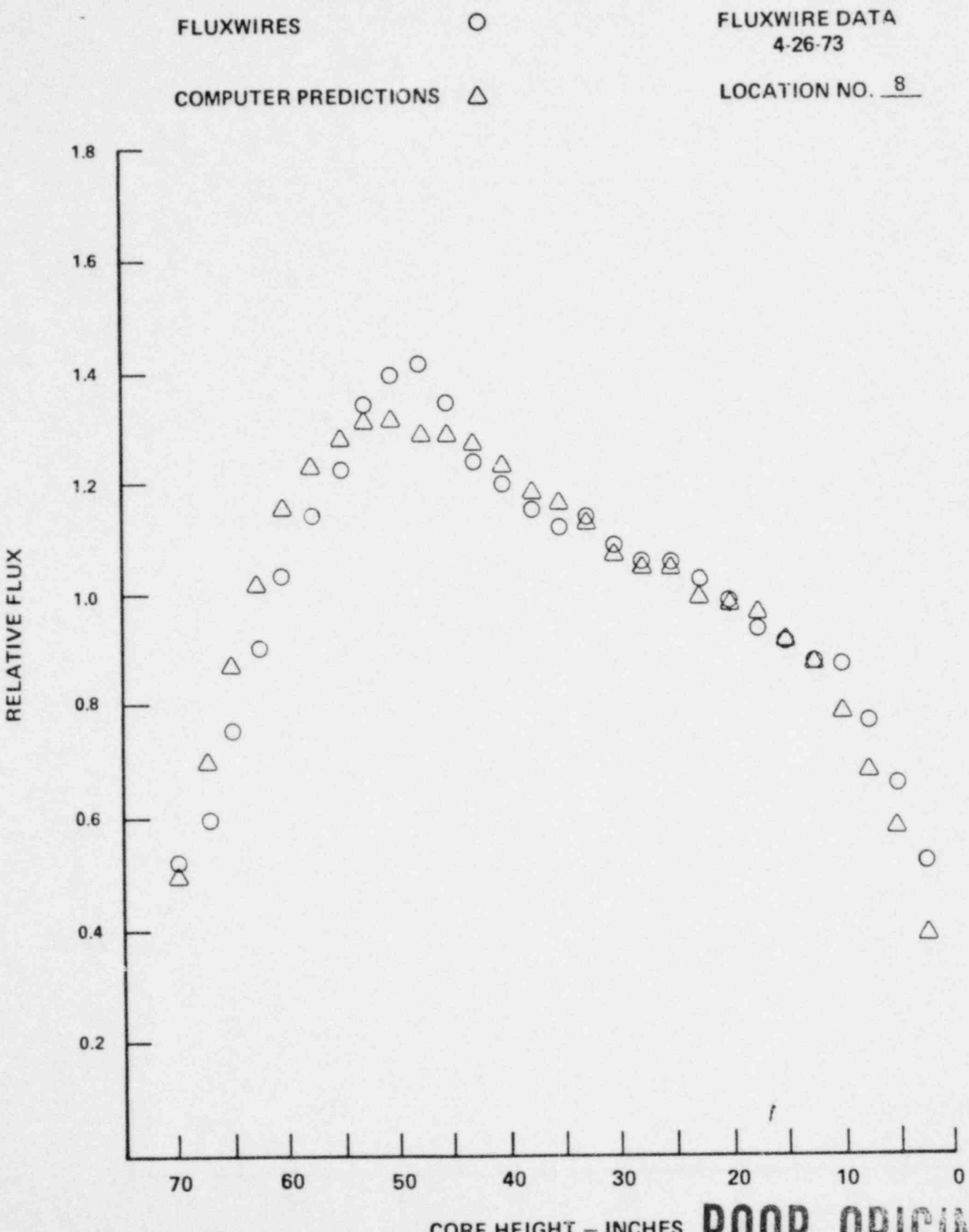


POOR ORIGINAL

FIGURE 10

NORMALIZED AXIAL PROFILE

BIG ROCK POINT



POOR ORIGINAL

FIGURE 11

BIG ROCK POINT

TEMPERATURE COEFFICIENT TEST – APRIL 6, 1973

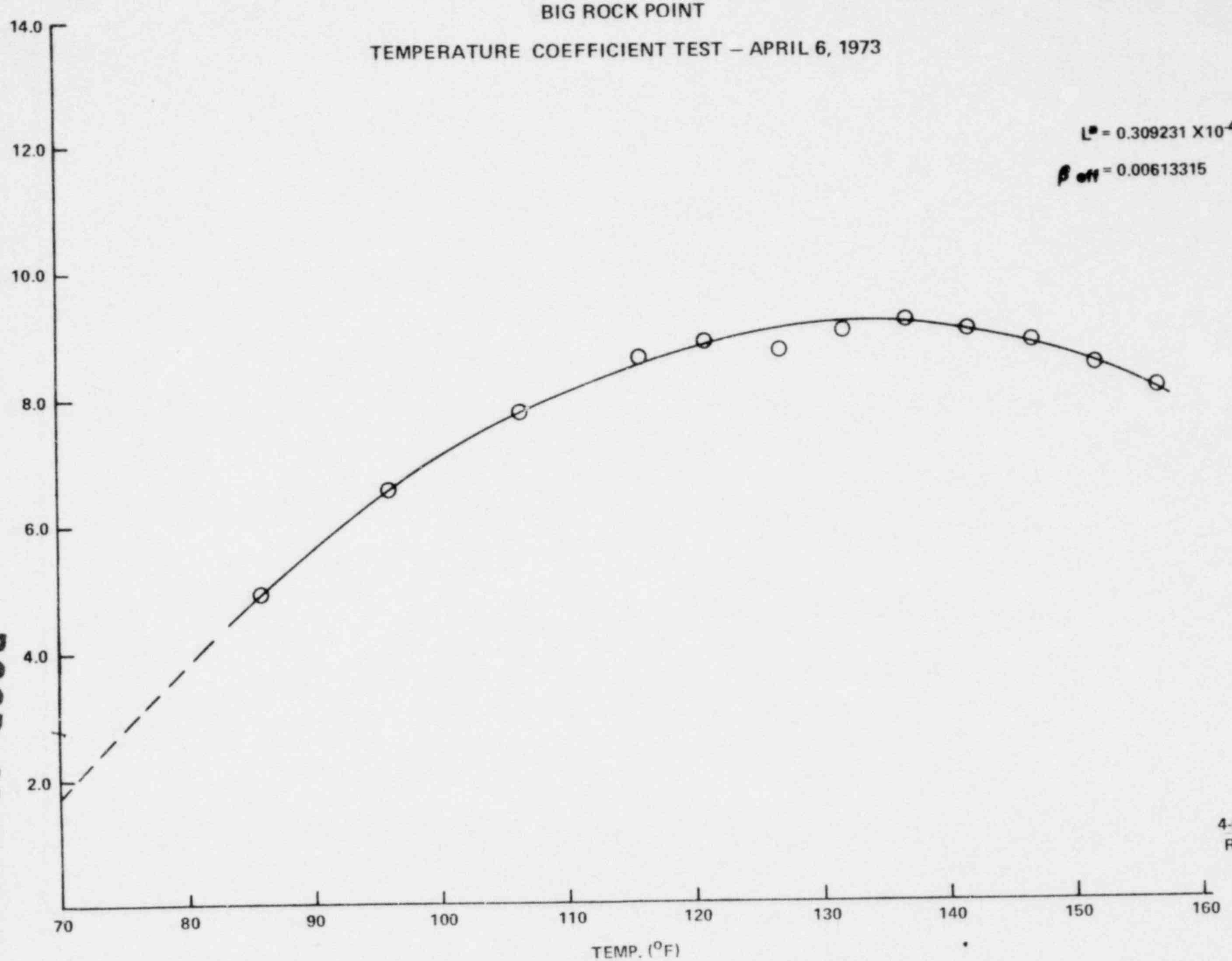


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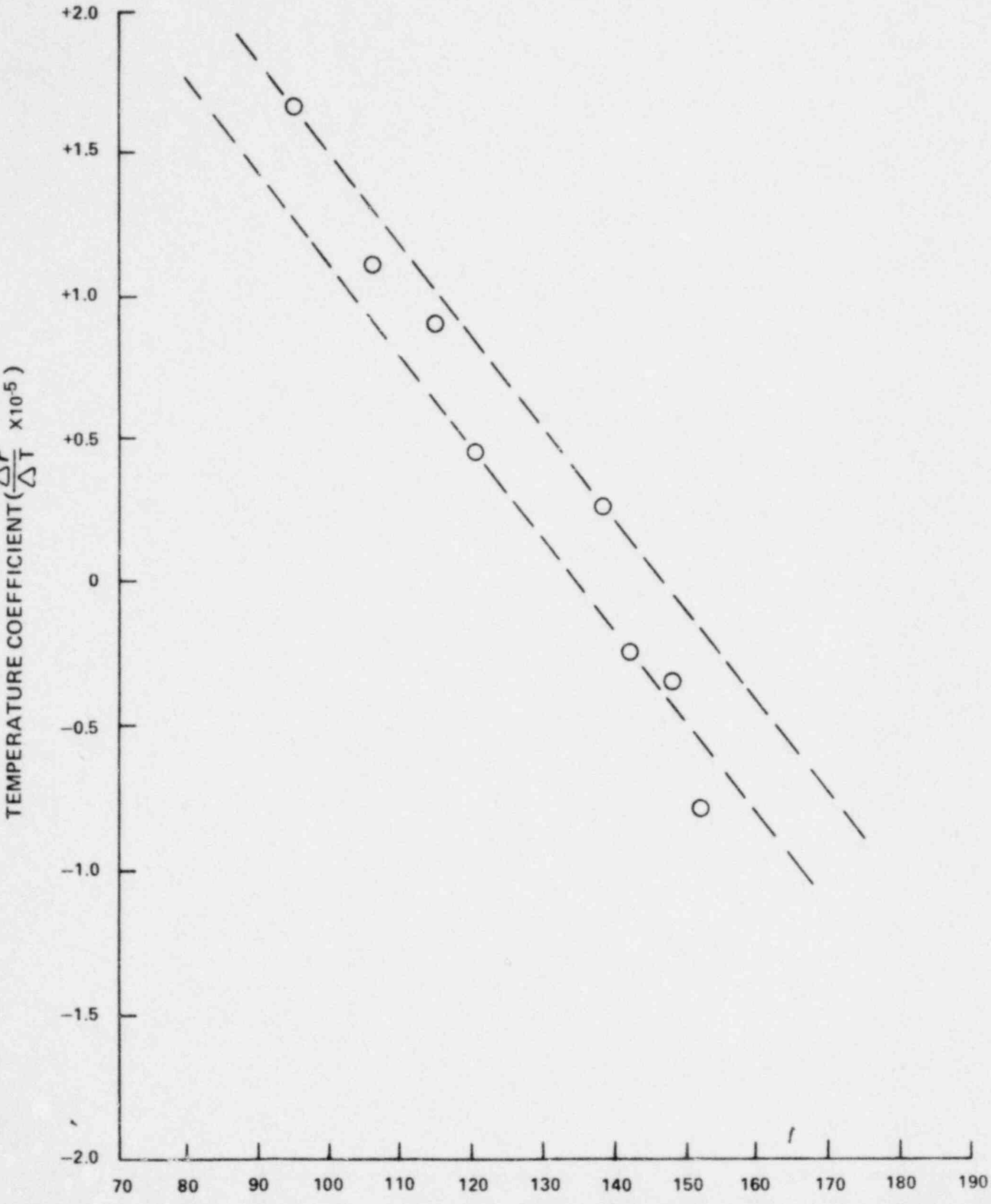
FIGURE 12

BIG ROCK POINT



POOR ORIGINAL

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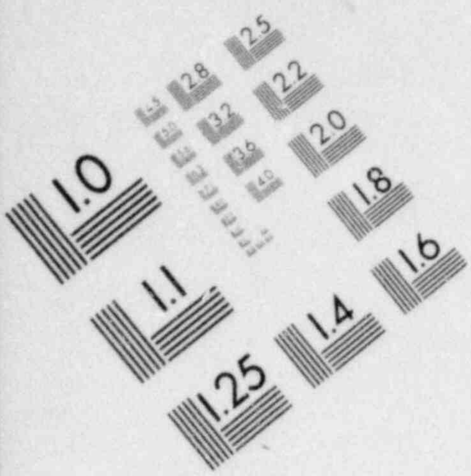
ENCLOSURE

SAMPLE TECHNICAL SPECIFICATION LANGUAGE

The following language should be substituted, as appropriate, into the Technical Specifications where existing surveillance requirements are superseded by ASME Section XI inservice inspection and testing requirements:

a. Inservice inspection of ASME Code Class 1, Class 2 and Class 3 components shall be performed in accordance with Section XI of the ASME Boiler and Presure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

b. Inservice testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).



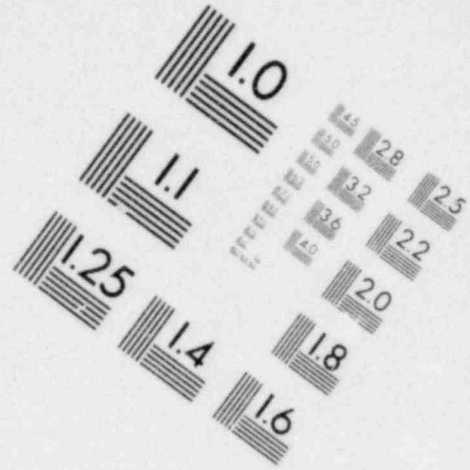
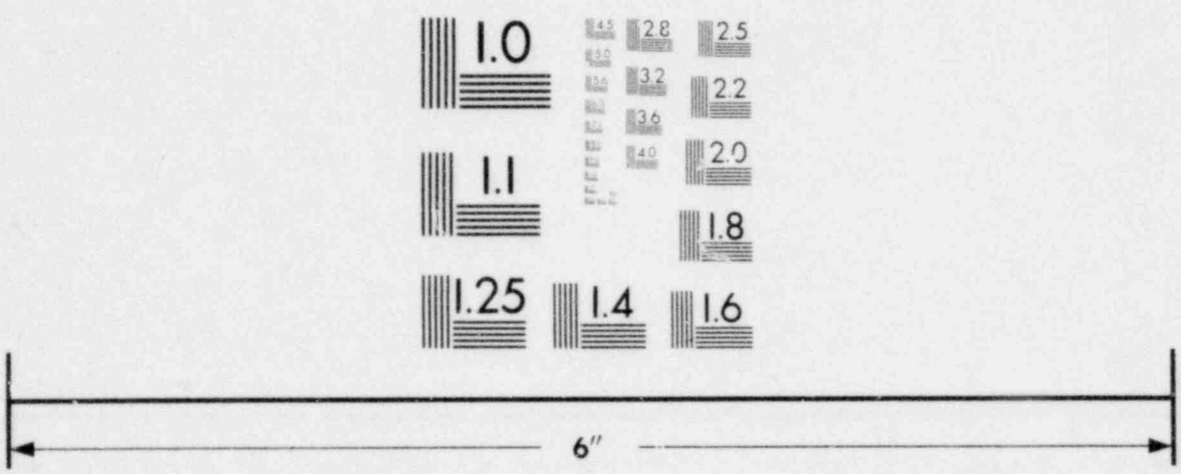
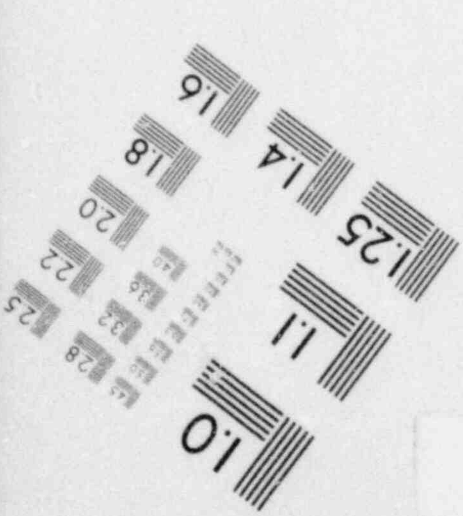


IMAGE EVALUATION TEST TARGET (MT-3)





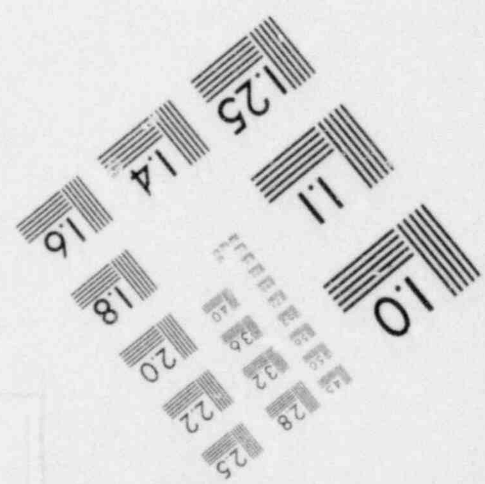
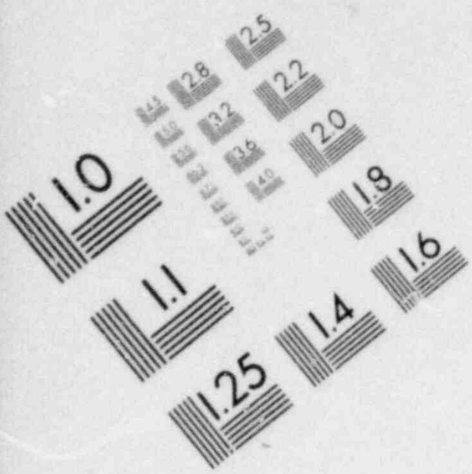
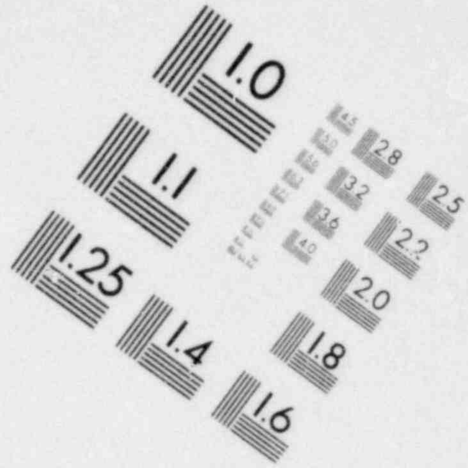
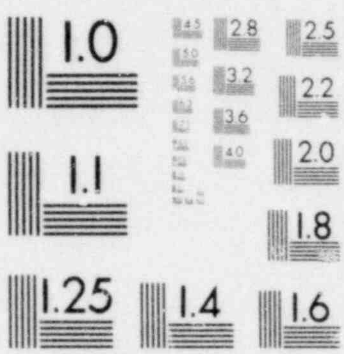


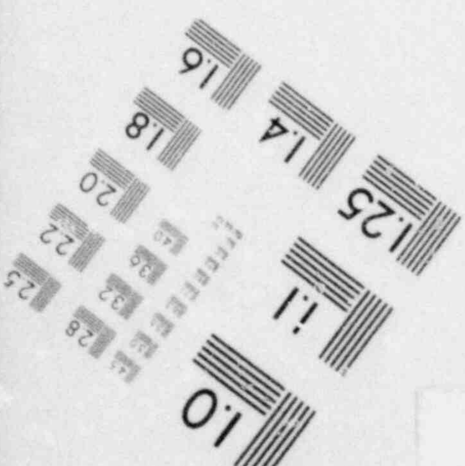
IMAGE EVALUATION
TEST TARGET (MT-3)

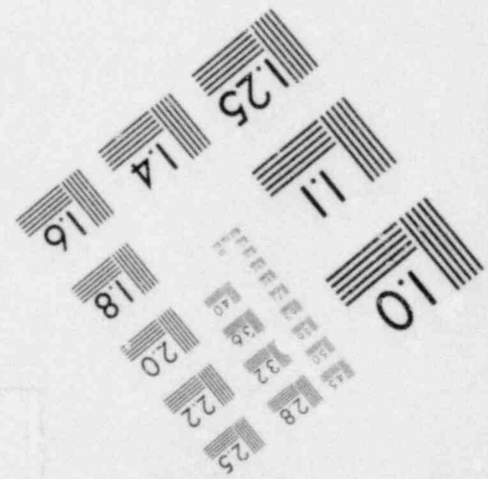






6″





(3) For construction permits issued on or after July 1, 1974, pumps which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code and Addenda[3,4] in effect[4] on the date of order[5] of the pump or 12 months prior to the formal docket date of the application for construction permit, whichever is later: *Provided*, That the applicable ASME Code provisions for pumps shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition. The pumps may meet the requirements set forth in subsequent editions of this Code and Addenda which become effective.

41 FR 6256

(f) Valves:

(1) For construction permits issued before January 1, 1971, for reactors not licensed for operation, valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements set forth in

(i) The American Standard Code for Pressure Piping (ASA B31.1), Addenda, and applicable Code Cases, or the USA Standard Code for Pressure Piping (USAS B31.1.0), Addenda, and applicable Code Cases, in effect[4] on the date of order[5] of the valves or the Class I section of the Draft ASME Code for Pumps and Valves for Nuclear Power,[3] Addenda, and Code Cases in effect on the date of order of the valves; or

(ii) The nondestructive examination and acceptance standards of ASA B31.1 Code Cases N2, N7, N9, and N10, except that the acceptance standards for Class I valves set forth in the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda in effect on the date of order of the valves may be applied.

The valves may meet the requirements set forth in editions of ASA B31.1, USAS B31.1.0, and the Draft ASME Code for Pumps and Valves for Nuclear Power, Addenda, and Code Cases, which became effective after the date of order of the valves.

37 FR 17021

(2) For construction permits issued on or after January 1, 1971, but before July 1, 1974,* valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class I valves set forth in editions of (i) the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[3] in effect[4] on the date of order[5] of the valves and the requirements applicable to valves set forth in articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda[3] in effect on the date of order of the valves, or (ii) the requirements applicable to Class 1 valves of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect on the date of order of the valve; *Provided, however*, That if the valves are ordered more than 12 months prior to the date of issuance of the construction permit, compliance with the requirements for Class I valves set forth in editions of the Draft ASME Code for Pumps and Valves for Nuclear Power and Addenda[3] and the requirements applicable to valves set forth in articles 1 and 8 of editions of section III of the ASME Boiler and Pressure Vessel Code and Addenda or for Class 1 valves of section III of the ASME Boiler and Pressure Vessel Code and Addenda in effect 12 months prior to the date of issuance of the construction permit is required. The valves may meet the requirements set forth in editions of these Codes or Addenda which have become effective after the date of valve order or after 12 months prior to the date of issuance of the construction permit.

37 FR 17021

(3) For construction permits issued on or after July 1, 1974, valves which are part of the reactor coolant pressure boundary[2] shall meet the requirements for Class 1 components set forth in Section III of the ASME Boiler and Pressure Vessel Code and Addenda[3,4] in effect[4] on the date of order[5] of the valve or 12 months prior to the formal docket date of the application for construction permit, whichever is later: *Provided*, That the applicable ASME Code provisions for valves shall be no earlier than those of the Winter 1972 Addenda of the 1971 edition. The valves may meet the requirements set forth in subsequent editions of this Code and Addenda which become effective.

(g) Inservice inspection requirements:

(1) For a facility whose construction permit was issued prior to January 1, 1971, components (including supports) shall meet the requirements of paragraphs (g)(4) and (g)(5) of this section to the extent practical. Components which are part of the reactor coolant pressure boundary[2] and their supports shall meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves shall meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3.

41 FR 6256

(2) For a facility whose construction permit was issued on or after January 1, 1971, but before July 1, 1974, components (including supports) which are classified as ASME Code Class 1 and Class 2 shall be designed and be provided with access to enable the performance of (i) inservice examination of such components (including supports) and (ii) tests for operational readiness of pumps and valves, and shall meet the preservice examination requirements set forth in editions of Section XI of the ASME Boiler and Pressure Vessel Code and Addenda in effect 6 months prior to the date of issuance of the construction permit. The components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which become effective.

(3) For a facility whose construction permit was issued on or after July 1, 1974:

(i) Components which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[3] applied to the construction of the particular component in accordance with paragraph (c), (d), (e), or (f) of this section.

(ii) Components which are classified as ASME Code Class 2 and Class 3 and supports for components which are classified as ASME Code Class 1, Class 2, and Class 3 shall be designed and be provided with access to enable the performance of inservice examination of such components and shall meet the preservice examination requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[3] applied to the construction of the particular component.

(iii) Pumps and valves which are classified as ASME Code Class 1 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[3] applied to the construction of the particular pump or valve in accordance with paragraphs (e) and (f) of this section or the Summer 1973 Addenda, whichever is later.

(iv) Pumps and valves which are classified as ASME Code Class 2 and Class 3 shall be designed and be provided with access to enable the performance of inservice testing of the pumps and valves for assessing operational readiness set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[3] applied to the construction of the particular pump or valve or the Summer 1973 Addenda, whichever is later.

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(v) All components (including supports) may meet the requirements set forth in subsequent editions of codes and addenda or portions thereof which become effective.

(4) Throughout the service life of a facility, components (including supports) which are classified as ASME Code Class 1, Class 2 and Class 3 shall meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda[3] that become effective subsequent to editions specified in paragraphs (g)(2) and (g)(3) of this section and are incorporated by reference in paragraph (b) of this section, to the extent practical within the limitations of design, geometry and materials of construction of the components.

(i) The initial inservice examination conducted during the first 40 months shall comply with the requirements in the editions of the code and addenda in effect no more than 6 months prior to the date of start of facility commercial operation.

(ii) The inservice examinations conducted during successive 40-month periods throughout the service life of the facility thereafter shall comply with

See page 50-14 for footnotes 1 through 6.
*Amended 41 FR 6256.

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those requirements in editio~ of the code and addenda in effect no than 6 months prior to the start of each 40-month period.

(iii) The initial inservice tests of pumps and valves for assessing operational readiness and system pressure tests conducted during the first 20 months shall comply with those requirements in editions of the code and addenda in effect no more than 6 months prior to the start of facility commercial operation.

(iv) Inservice tests of pumps and valves for assessing operational readiness and system pressure tests conducted during successive 20-month periods throughout the service life of the facility shall comply with those requirements in editions of the code and addenda in effect no more than 6 months prior to the start of each 20-month period.

(v) For an operating facility whose operating license was issued prior to March 1, 1976, the provisions of paragraph (g)(4) of this section shall become effective after September 1, 1976, at the start of the next regular 40-month period of a series of such periods beginning at the start of facility commercial operation.

(5)(i) The inservice inspection program for a facility shall be revised by the licensee, as necessary, to meet the requirements of paragraph (g)(4) of this section.

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(ii) If a revised inservice inspection program for a facility conflicts with the technical specification for the facility, the licensee shall apply to the Commission for amendment of the technical specifications to conform the technical specification to the revised program. This application shall be submitted at least 6 months before the start of the period during which the provisions become applicable as determined by paragraph (g)(4) of this section.

(iii) If the licensee has determined that conformance with certain code requirements is impractical for his facility, the licensee shall notify the Commission and submit information to support his determinations.

(iv) Where an examination or test requirement by the code or addenda is determined to be impractical by the licensee and is not included in the revised inservice inspection program as permitted by paragraph (g)(4) of this section, the basis for this determination shall be demonstrated to the satisfaction of the Commission not later than 12 months after the expiration of the initial 120-month period of operation from start of facility commercial operation and each subsequent 120-month period of operation during which the examination or test is determined to be impractical.

(6)(i) The Commission will evaluate determinations under paragraph (g)(5) of this section that code requirements are impractical and may grant such relief as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if requirements were imposed on the facility.

(ii) The Commission may require the licensee to follow an augmented inservice inspection program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

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(h) Protection systems: For construction permits issued after January 1, 1971, protection systems shall meet the requirements set forth in editions or revisions of the Institute of Electrical and Electronics Engineers Standard: "Criteria for Protection Systems for Nuclear Power Generating Stations," (IEEE-279) in effect on the formal docket date of the application for a construction permit. Protection systems may meet the requirements set forth in subsequent editions or revisions of IEEE-279 which become effective.