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LTR 3 ENCL 40

SUBJECT: RESPONSE TO NRC LTR DTD 05/02/78... FORWARDING RESPONSE TO QUESTIONS 1, 2, 3C,  
5, 6, 7, 8, & 10 CONCERNING CONTINUED CYCLE 1 OPERATION WITHOUT BURNABLE  
POISON ROD ASSEMBLIES... NOTARIZED 06/02/78.

PLANT NAME: CRYSTAL RIVER #3

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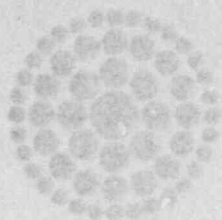
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**Florida  
Power**  
CORPORATION

June 2, 1978

Mr. Robert W. Reid, Chief  
Operating Reactors Branch #4  
Division of Operating Reactors  
U. S. Nuclear Regulatory Commission  
Washington, D.C. 20555

Subject: Crystal River Unit #3  
Docket No. 50-302  
Operating License #DPR-72

REGULATORY DOCKET FILE COPY

Dear Sir:

On May 8, 1978 Florida Power Corporation received your letter of May 2, 1978 requesting us to supply additional information in response to the questions contained in Enclosures 1 and 2 of your letter.

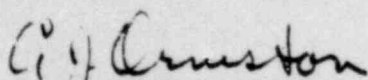
The responses to question 9 of Enclosure 2 and questions 1 through 9 of Enclosure 1 were submitted to the Commission on May 12, 1978 and May 16, 1978, respectively.

Attached is Florida Power Corporation's responses to questions 1, 2, 3c, 5, 6, 7, 8 and 10 of Enclosure 2 of your May 2, 1978 letter. Our responses to questions 3a, 3b, 3d, 3e, and 4 are contained in the B&W report entitled "BAW-1490, Crystal River Unit 3, Licensing Considerations for Continued Cycle 1 Operation Without Burnable Poison Rod Assemblies," dated May 1978. This B&W report is being submitted via a separate letter to you on this date.

This filing of information completes Florida Power Corporation's response to your request for additional information contained in your May 2, 1978 letter. Should you or members of your staff have any questions or comments regarding our submittals, please feel free to contact this office.

Very truly yours,

FLORIDA POWER CORPORATION

  
A. J. Ormston  
Vice President

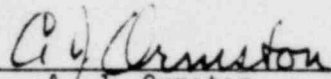
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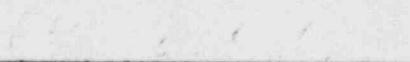
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STATE OF FLORIDA  
COUNTY OF PINELLAS

A. J. Ormston, Vice President for Florida Power Corporation, states that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information and belief.

  
A. J. Ormston

Subscribed and sworn to before me, a Notary Public in and for the State and County above-named, this 2nd day of June, 1978.

  
Notary Public

Notary Public, State of Florida at Large,  
My Commission Expires: July 25, 1980

Question 1 - Describe how any assymetries in the core exposure distribution will be accounted for in predictive core physics calculations and in the incore monitoring routine.

Answer:

The majority of cycle 1 operated with nearly zero quadrant tilt as measured by the incore monitoring system. The escape of the two Burnable Poison Rod Assemblies (BPRAs) from the core at approximately 200 EFPD caused only a minor perturbation to the symmetric exposure distribution. A maximum burnup difference of  $\sim 0.4\%$  occurred in the fuel assemblies with the BPRAs removed. This small local effect is not expected to perturb the core physics parameters calculated for a symmetric burnup distribution.

Question 2 - Describe how the uncertainty in the exposure of the two fuel assemblies from which the burnable poison clusters were uncoupled will be accounted for in predictive calculations and in incore monitoring.

Answer:

The error in the assembly exposure calculated by the plant computer has been estimated to be no greater than 0.4% for the two assemblies from which burnable poison clusters were uncoupled. An error of this magnitude will not significantly impact the signal-to-power conversion by the incore monitoring system during the remainder of cycle 1 operation.

Question 3c- Provide a safety analysis for operation with the burnable poison removed. This analysis should address the effects on axial peaking factors, "flyspeck" analyses and axial imbalance limits. Give quantitative information on the axially-dependent depletion of burnable poison.

Answer:

The effect of BPRAs removal on the axial power distribution is shown in Figures 1 and 2. The removal of BPRAs causes a shift in the overall distribution, however the magnitude of the axial peak is relatively unchanged. A complete power peaking analysis for the remainder of cycle 1, with BPRAs removed, was performed for C. 3. The study included normal operating conditions, abnormal core conditions including mispositioned control rods and APSRs, and variable xenon conditions. The resultant power peaks were compared to the thermal limits, DNBR and centerline fuel melt, by means of "flyspeck" curves. The original axial imbalance limits for the Reactor Protection System were determined to be conservative for the remainder of cycle 1 with BPRAs removed. The axial imbalance limits for LOCA power peaking have been altered for the remainder of cycle 1 as shown in Reference 3. The LOCA peaking limits are a function of axial position. The removal of BPRAs alters the overall axial distribution such that the peaking behavior at various axial positions is different from the original design.

The burnable poison (isotope B-10) depletes non-uniformly in the axial direction. Early in core life (BOC) the axial flux shape approximates a cosine shape, Figure 3. Since the flux is higher at the midplane, the B-10 there depletes faster than at the extremities. This causes an increase in the axial peak at ~50 EFPD as shown in Figure 4. Continued operation of the core leads to a more rapid fuel depletion at the midplane - because of the higher power. The resulting power shape at 254 EFPD, Figure 1, is the relatively flat distribution typical of a depleted core. The axial shapes for the three times in core life are plotted together in Figure 5 which summarizes the combined effects of fuel and poison depletion.

At this time the B-10 is nearly fully depleted. However, the residual B-10 is greater at the extremities and in effect "pushes" power toward the midplane. When the BPRAs are removed the power will shift away from the depleted midplane toward the extremities as indicated in Figure 2.

Question 5 - Describe quantitatively the methods which will be used to detect further failing (eg flux tilts incore maps, cleanup system, loose parts monitor). In particular, to what impact energy, on the reactor vessel head and steam generator tube sheet and upper head, do the loose-part monitor alarm setpoints correspond?

Answer:

As all BPRAs are being removed from the CR-3 core, future failures are impossible. Loose parts in general will be detected by the Loose Parts Monitor. Two sensors on the reactor vessel head and two on each OTSG upper tubesheet will be set to alarm for an impact energy of 0.5 ft. lb at a minimum distance of 3 feet. This setpoint is in agreement with NRC Regulatory Guide 1.133.

Note:

Questions 6, 7, 8 are answered in unison as they all pertain to Once-Through-Steam Generators (OTSG's). The answer takes the form of an updated status of the OTSG test and repair program.

Question 6 - Describe the damage to the steam generator tubesheets and steam generator tubes.

Question 7 - Describe your plans to repair the damaged components of the steam generators including the procedures for rewelding grinding or milling, testing, and inspection.

Question 8 - Provide the results of the inspections, tests, and repairs discussed above.

Answer:

B&W has concluded that combined operation of CR III "B" OTSG for a short period (1 year or less) does not represent a safety problem. The damage has been limited to tube ends and weld deposited Inconel. The strength of the tube-to-tubesheet joint to resist an accident event has not been reduced since the "hard" tube roll strength capacity has been determined independent of the seal weld. The seal weld minimum leak path has not been decreased below minimum ASME code requirements for a seal weld, the seal capabilities of the weld for the short term have only been effected in the areas of low cycle fatigue and stress corrosion cracking. Low cycle fatigue is not considered a safety problem since only plant heat-ups and cooldowns affect this property. Failures of this nature will only cause plant shutdowns for repair when the leakage exceeds plant limitations. Stress corrosion cracking is a long term phenomenon requiring incubation periods longer than the anticipated time to the next refueling and planned generator restoration. Again failures of this nature result only in plant shutdowns for weld repairs when leakage exceeds plant limitations.

#### Background

The structural attachment of the tube to the tubesheet is at the primary side of the tubesheet. The 56 foot long, 5/8" O.D. tubes extend from the primary sides of the upper and lower tubesheets, through the two foot thick tubesheets, and through 15 support plates spaced three to four feet apart. The attachment to the tubesheet consists of a "hard" roll expansion approximately 1 inch into the tubesheet measured from the tube-to-tubesheet fillet seal weld on the tubesheet primary face.

A high quality automatic machine fillet weld at the tubesheet primary face provides the primary-to-secondary side seal. This is an Inconel weld between the Inconel tube and Inconel clad (5/16" thick) tubesheet.

Loads that are imposed on the tube-to-tubesheet joint are primarily axial loads due to thermal differentials and pressure fluctuations (pulses). Lateral loads and their resulting moments applied by secondary side conditions are attenuated within the 23 inches of tubesheet before reaching the 1 inch "hard" tube roll.

### Damage

Damage to the tube end extending above the tube-to-tubesheet weld, the weld itself, and clad has been varied. The tube-to-tubesheet damage has been classified into the following categories:

Class I (55% of the tubes)

Impact or roll over of tube ends may exist on the O.D. or I.D. Deformed material does not include weld metal.

Class II (6% of the tubes)

Partially separated chip (silver); may exist with Class I, III, or IV damage.

Class III (26% of the tubes)

Minor weld damage extending into the upper 1/3 of weld metal.

Class IV (17% of the tubes)

Damage to the tube ends and weld metal is excess of Class III. (Above percentages exceed 100% since Class II can exist with Class I, III, & IV).

### Calculations

Calculations of potential leakage rates were made assuming various widths of tube-to-tubesheet leakage paths. The paths were conservatively assumed to exist 360° around the tube.



For an 0.5 mil gap (a conservatively large crack width) the leakage was estimated to be 0.2 gpm with normal RCS operating pressure and zero secondary system pressure (this worst case differential pressure simulates steam line break conditions). Further, if all of the tubes with significant (Class IV) weld damage were to leak simultaneously, the total leakage would be 550 gpm.

The worst case leakage calculated above for Crystal River 3 is approximately equal to the leak rate associated with a double-ended tube break concurrent with a steam line break. Environmental radiation releases based on the double-ended tube break and concurrent steam line break were calculated for the Oconee plant (Docket 50269-900) and the resulting dose consequences at the site boundary were about 5% of the 10 CFR 100 limits. Based on the similarity of the plant designs and the difference in site meteorology conditions (Oconee  $\lambda/Q$  of  $1.16 \times 10^{-4} \text{ sec}/\text{m}^3$  versus CR-3  $\lambda/Q$  of  $1.78 \times 10^{-4} \text{ sec}/\text{m}^3$ ), it can be concluded that the CR-3 two-hour site boundary doses resulting from the release of radioactivity from the damaged tube-to-tubesheet welds under steam line break conditions would be less than 10% of the 10 CFR 100 limits.

### Tests

A ten tube mock-up of the damage was prepared at B&W fabrication facilities. The mock-up damage was made by beating the tubes with a ballpeen hammer. The damage mock-up includes the worst damage observed at the site. Hardness traverses across the damaged tube ends, welds, and into the clad showed the effects of significant cold working. The highest of the 250 readings were mostly in the high 300's (Knoop hardness scale) with three readings exceeding 400. Normal readings for Inconel weld metal may range as high as 200. The high hardness readings indicate increased yield and ultimate strengths, potentially increased resistance to high cycle fatigue failure, decreased resistance to low cycle fatigue failure, and decreased resistance to stress corrosion cracking.

The minimum leak path (normally the weld throat of the the tube-to-tubesheet fillet weld) of the mock-up damage was measured. Measurements were made per procedures used to inspect daily product quality control samples of shop production welds (stage micrometer at approximately 15 x power). The minimum measured leak path was .051 inches compared to minimum tube wall thickness of 0.032 inches.

Tests of the roll joint showed a minimum strength of 2500 pounds axial tube load to initiate relative motion with the tubesheet. The minimum load to completely free the tube from the tubesheet was 4520 lbs. The minimum tube yield strength based on the specified minimum yield stress of 35,000 psi is 2200 lbs. The tests were performed on samples that were expanded, welded, and stress relieved per standard B&W fabrication procedures. The welds were removed in their entirety prior to performing the tube pulling tests.

#### Repair

Repair of the steam generator has been separated into two phases: a short term repair and a long term repair. The short term is presently underway and consists of:

- Video Inspection & Cat. Tubes
- Locate Leaks & Identify
- Install Dome Shielding & J Leg Screens
- Repair Leaks, NDE Repairs, & Dress Tubes
- Remove Shielding
- 100% Free Path Check
- Clean Obstructed Tubes & E/C
- Explosive Plug & Leak Test
- 100% Free Path Check OTSG-A
- Remove Screens & Close Up OTSGs

The video inspections and cataloging of damage has been completed and the results reported above. All tube-to-tubesheet welds have been inspected and no leaks have been found.

Leak testing was performed by pressurizing the partially filled secondary side with helium and inspecting each weld individually with a mass spectrometer capable of detecting a  $10^{-8}$  cc/sec leak. The dome shielding has been installed and the tube ends dressed with hand held tools. The remaining items of the short term repair have yet to be completed.

The following long term repair will be performed at the end of the first fuel cycle:

- Dome Decon Prep.
- Install Decon Equip. Bladders
- Decontaminate Dome
- Setup Machine Carriage
- Machine/Spot Face Tubes (57%/43%)
- Deburr Tube Ends & Vacuum
- Clean Spot Face As Necessary
- Roll Expand as Necessary
- Remove Mach. Carriage
- Setup & Preheat Tubesheet
- Install/Setup Weld Carriage
- Weld Spot Face Tubes (2800)
- Video Inspect Welds
- Clean for Pt Inspection

Remote machines to support the long term repair are presently being designed and fabricated. Prior to installation at the site, all remotely controlled machine will be proven in mock-ups.

### Conclusion

Based on the above assessment the structural capacity of the tube-to-tube-sheet joint has not been degraded because of the "hard" roll capacity to continue to carry axial tube loads greater than the tube yield strength and the attenuation of lateral tube loads up through the 2 foot thick tubesheet, Therefore the damage sustained by the OTSG will not have a detrimental effect on its safe operation over the time remaining on this fuel cycle.

Figure 1  
CRYSTAL RIVER III CYCLE 1  
254 EFPD

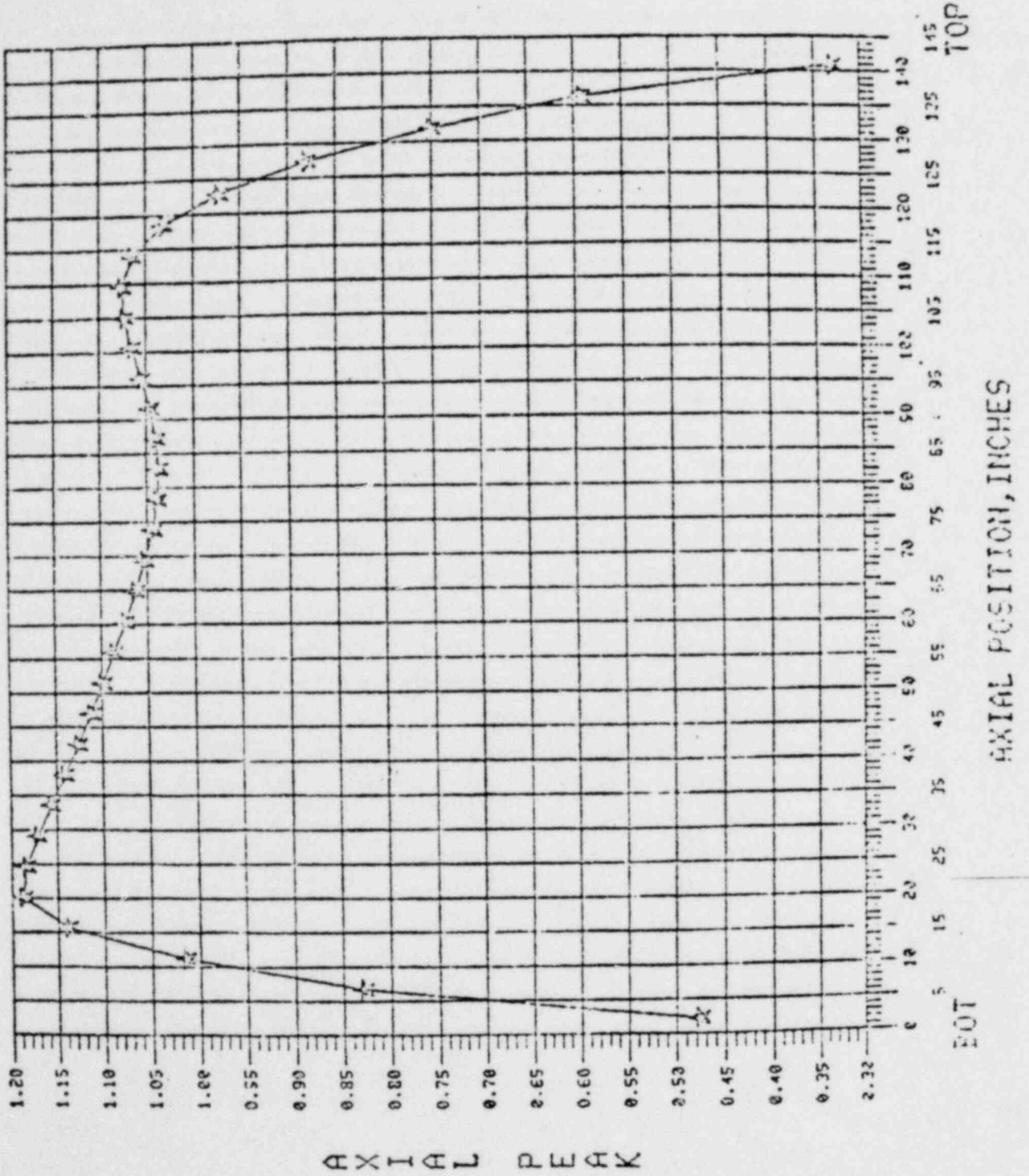
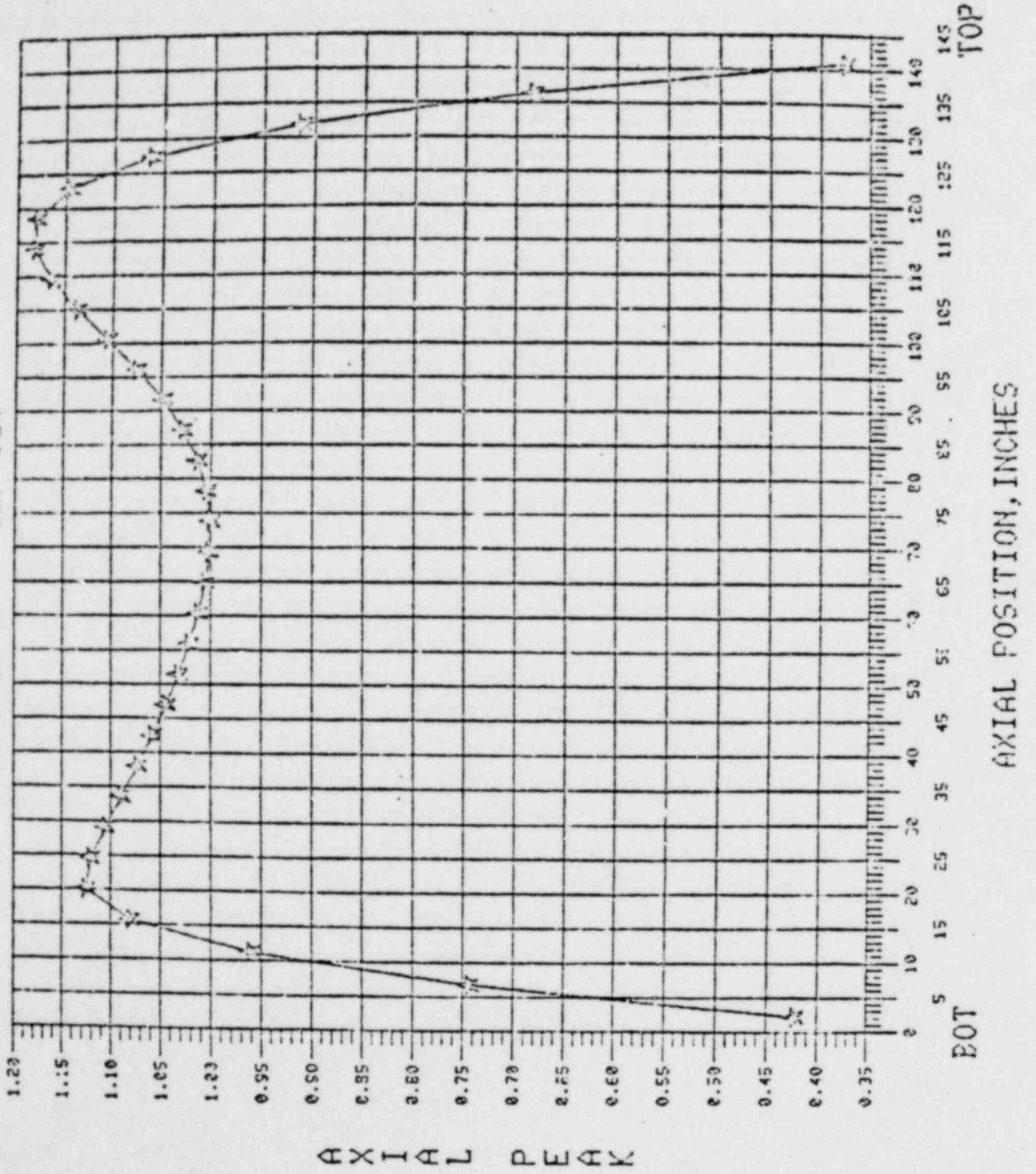


Figure 2

CRYSTAL RIVER III CYCLE 1

273 EFPD LBP OUT



Question 10 - Provide an estimate of all offsite releases which may result from the repair effort. Such releases may include liquid waste from decontamination solutions or airborne particulates from grinding. Indicate what treatment systems will be used to reduce levels of radioactivity in plant effluents.

Answer:

The estimated off-site releases which may result from the steam generator repair effort are as follows:

Gaseous - No increase in airborne activity releases are expected due to installation of temporary absolute and charcoal filtration systems at the steam generator gaseous outlets.

Liquids - Existing waste treatment systems, such as evaporation for volume reduction, may be supplemented by a vendor-supplied demineralization unit. It is anticipated that approximately 750,000 gallons will require processing which will result in an estimated off-site release of <0.5 Ci.

Solids - It is anticipated that the following solid waste will be generated:

- 1) Compacted wastes - ~1800 ft.<sup>3</sup>, <6 Ci
- 2) Solidified evaporator bottoms and spent resins - 10,000 ft.<sup>3</sup>, ~5 Ci

Figure 3

CRYSTAL RIVER III CYCLE 1

4 EFPD

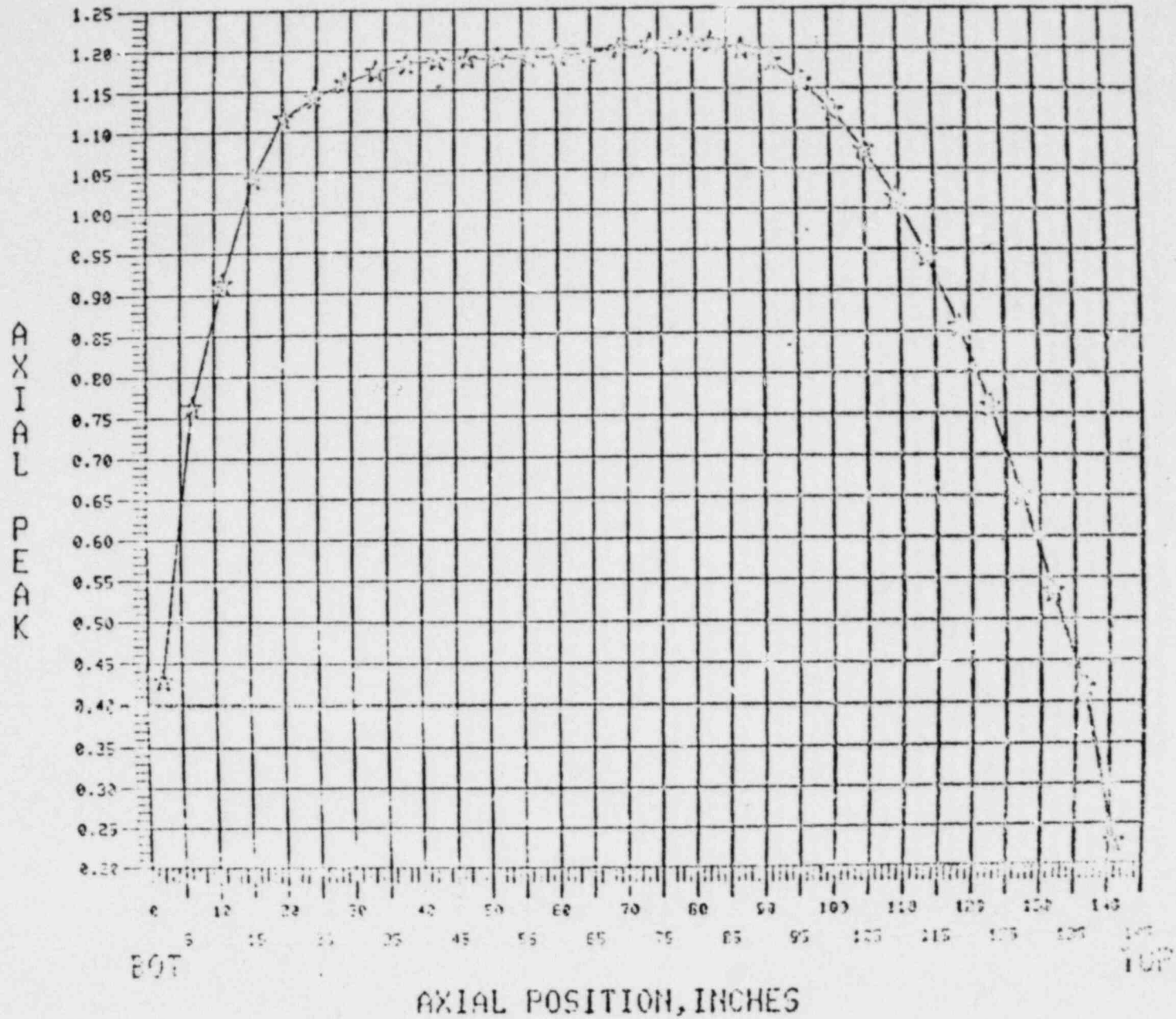
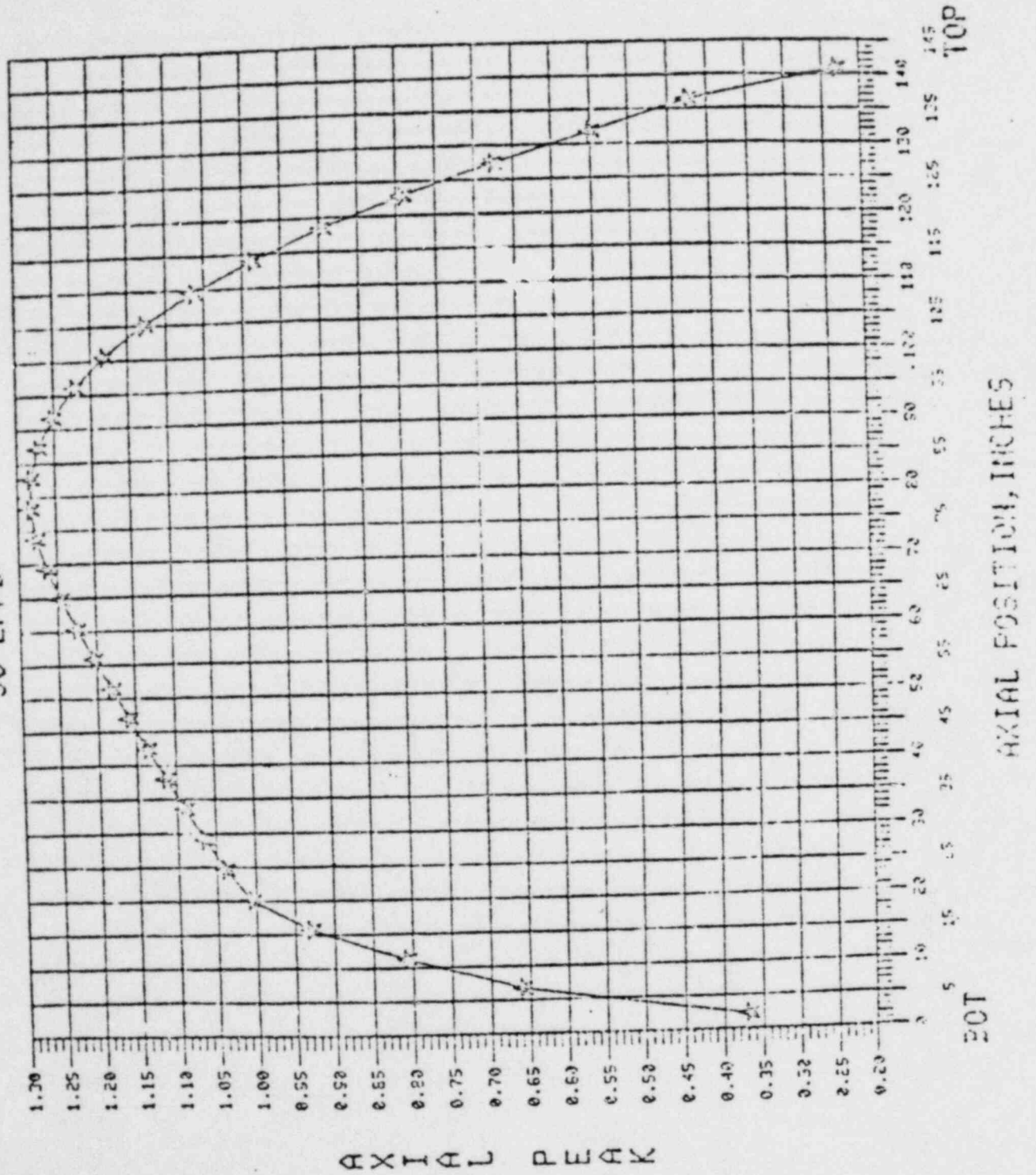


Figure 4

CRYSTAL RIVER III CYCLE 1

50 EFPD



20T

TOP



Figure 5

CRYSTAL RIVER III CYCLE 1

254 EFPD-TRI.    4 EFPD-DIA.    50 EFPD-STAR

