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SUBJECT:
RESPONSE TO NRC LTR DTD 05/02/78...FORWARDING ADDL INFO CONCERNING THE FAILURE OF BURNABLE POISON ROD ASSEMBLIES AT UNIT 3.

PLANT NAME: CRYSTAL RIVER #3

REVIEWER INITIAL: XJM
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Florida Power
CORPORATION

W. P. STEWART, DIRECTOR
POWER PRODUCTION

May 12, 1978

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

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

Dear Sir:

On May 8, 1978 we received Mr. Reid's letter of May 2, 1978 requesting additional information concerning the failure of burnable poison rod assemblies (BPRA) at Crystal River Unit 3.

As requested, Florida Power Corporation hereby formally submits our response to Question No. 9 of Enclosure 2 of Mr. Reid's letter. This information was informally transmitted to the Commission for review on May 9, 1978.

Our schedule for transmitting the remaining requested information as well as our schedule for core reload, steam generator repairs, and reactor restart will be submitted on May 15, 1978.

If you or members of your staff have any questions concerning our response, please contact this office.

Very truly yours,

FLORIDA POWER CORPORATION

O. B. DuBois / for

W.P. Stewart

WPS/ECS/mcm5/2a

File: 3-0-3-a-3

A001 S 1/40

General Office 3201 Thirty-fourth Street South • P.O. Box 14042, St. Petersburg, Florida 33733 • 813—866-5151

Exposure Plan for CR-3 Steam Generator Repairs

The Crystal River-3 steam generator (SG) repair effort will be performed such that radiation exposure is as low as reasonably achievable (ALARA), consistent with the intent of Regulatory Guides 8.8 and 8.10. Planning, mockups, training and shielding will be used to reduce exposure. Decontamination was extensively evaluated and judged not viable for this repair effort. However, the CR-3 plant staff utilized a demineralized water rinse which provided an effective decontamination of the steam generator. Due to its solvent characteristics this method of decontamination provided a D.F. of 1.7. The detailed evaluation follows.

Background

The repair plan was formulated after an extensive inspection of damaged areas. No damage was found in the A SG. Loose parts-induced damage was found in the B SG upper head; no damage was found in the lower head. Each tube end and weld in the upper head was videotaped using remote techniques, and the videotapes were examined to determine damage extent.

Table 1 shows the work steps for SG repair, and the estimated time required in the SG for each task. These times are conservative estimates by personnel experienced in SG field repair work; proper training in mockups before task performance is assumed. Figure 1 shows unshielded radiation levels in the B SG upper head.

The major contribution to exposure is work inside the B SG upper head, so a detailed analysis was performed to determine methods to reduce exposure in this area.

The radiation source inside the SG head is activated corrosion products plated on the primary side surfaces. Analyzed crud samples from this area (corrected for decay to May 1978) show the isotopic distribution of the crud is 74% Co-58, 17% Cr-51, and 9% Co-60.

For analysis, the sources inside the SG head were separated into contributions from the tubes, tubesheet, dome, and inlet nozzle. Contributions to the dose at several points inside the head were calculated. The dose two feet above the tubesheet, along the axial conterline and 3.42 feet off the centerline, generally represents the average dose received by workers inside the head during repair efforts. Shielded probe measurements made inside the head defined source term distribution from the four surfaces. Combining the measured source distribution with the calculated contribution, the dose contribution from the four surfaces are as follows:

Source	Axial Centerline	3.42' from Axial Centerline
Tubes	32%	34%
Tubesheet	33%	24%
Dome	25%	38%
Nozzle	10%	4%

Decontamination

Decontamination was evaluated in detail and was discussed with major decontamination vendors.

Full chemical decontamination (fill-soak-drain) is not viable alternative because:

a. Solvents for use with sensitized Inconel (SG tubes) are not fully qualified

b. Up to 150,000 gallons of high and low-level radwaste would be generated

c. Decontamination could not be completed before December 1978.

Evaluation showed fill-soak-drain decontamination risks outweighed the potential benefits.

Grit-blasting and vacuuming were considered. Irregular surfaces, such as the tubesheet, are difficult to clean by grit-blast. The dome and inlet nozzle surfaces are cleanable by grit-blast. The device used must be capable of grit retrieval to preclude grit contamination of the reactor coolant system. A suitable vendor was contacted who estimated the cleaning rate in the SG head at 10 square feet per hour. Also the grit-blast nozzle must operate within a one-inch band to assure proper grit pickup. The narrow operating band renders remote operation unreliable. The nozzle must be manually handled by personnel inside the SG to grit-blast the dome and nozzle; this requires about 50 rem exposure.

Grit-blasting experience at other nuclear plants shows that airborne contamination control is extremely difficult during grit-blasting. Further, disposal of the grit and corrosion products would require an estimated 2-5 manrem.

Therefore, grit-blasting is not considered viable because of the high dose required for the operation and the difficulties of maintaining proper airborne contamination control during the job.

Remote Operations

Remote tooling was considered for all tasks. Remote tooling will be used for the tube free path check and to clear obstructed tubes where practical.

Training

Each worker will be trained on the specific task to be performed prior to performing said task. A mockup of the upper SG head and dome has been setup at the site to allow detailed training under actual work conditions. During training, workers will be dressed in the Anti-C clothing required for actual job performance. The mockup will also be used to determine the most efficient steps for task performance, and workers will be trained in these steps. Training in the mockup will reduce required exposure by at least a factor of two, and possibly by a factor of three to four.

Each worker will also have training in site procedures and general radiation area work habits. Site Health Physics personnel will closely supervise all work performed in high radiation areas to assure proper exposure minimization procedures are followed.

Planning

As previously mentioned, an extensive inspection has been performed to determine repair tasks. Personnel experienced in radiation control techniques actively participated in deriving the detailed work plan. All tasks are designed to minimize exposure.

Shielding

Shielding was evaluated for each work task. Shielding is most effective for tasks, inside the OTSG's, which cannot be performed remotely. Exposures required for shielding installation and removal were calculated. Exposure required for each task, with and without shielding, were calculated. Portable shielding saves about 99 man-rem for repair tasks inside the SG's.

All shielding used inside the SG's must pass through the 16-inch I.D. access manway. Blankets, consisting of lead wool encased in Herculite, are commercially available, can be folded to pass throught the manway, and are easily handled. Each blanket provides the equivalent of 1/6 inch bulk lead shielding. These blankets will be used for temporary shielding.

Dose contributions from surfaces inside the SG head were previously given. Averaging the contributions gives the following source term importance two feet above the tubesheet:

Tubes and tubesheet	62%
Dome and inlet nozzle	38%

The isotopic source distribution was previously given. Theory predicts that two lead blankets will reduce the dose by 60%, for this isotopic distribution. Experiements in the SG show larger reductions, probably due to some beta in the field and lesser scattering for lead than iron. A dose reduction factor of 70%, for two lead blankets is consistent with the experimental data, and was used for shielding calculations.

Upper tubesheet shielding is easily installed; a detailed evaluation is not necessary to justify its use, except where remote operations are precluded by tubesheet shielding. A support structure to support lead blankets (two thick) over the dome has been designed, and evaluated for use. The average dose rate (unshielded) two feet above the tubesheet is 3.0 to 3.5 R/hr. Calculated dose rates, based on the previously discussed shielding model and shielding attenuation, are given below:

Unshielded	3.5 R/hr
Tubesheet Shielding	2.0 R/hr
Dome Shielding	2.5 R/hr
Tubesheet plus Dome Shielding	1.0 R/hr

Table 2 shows exposures for the repair tasks inside B SG upper head. Tubesheet shielding saves about 65 rem, and dome shielding saves an additional 29 rem. for B SG upper head repair.

Dome shielding installation is currently being checked out in the mockup, to assure that installation time will not exceed four manhours. Both dome and tubesheet shielding is planned during upper head repair.

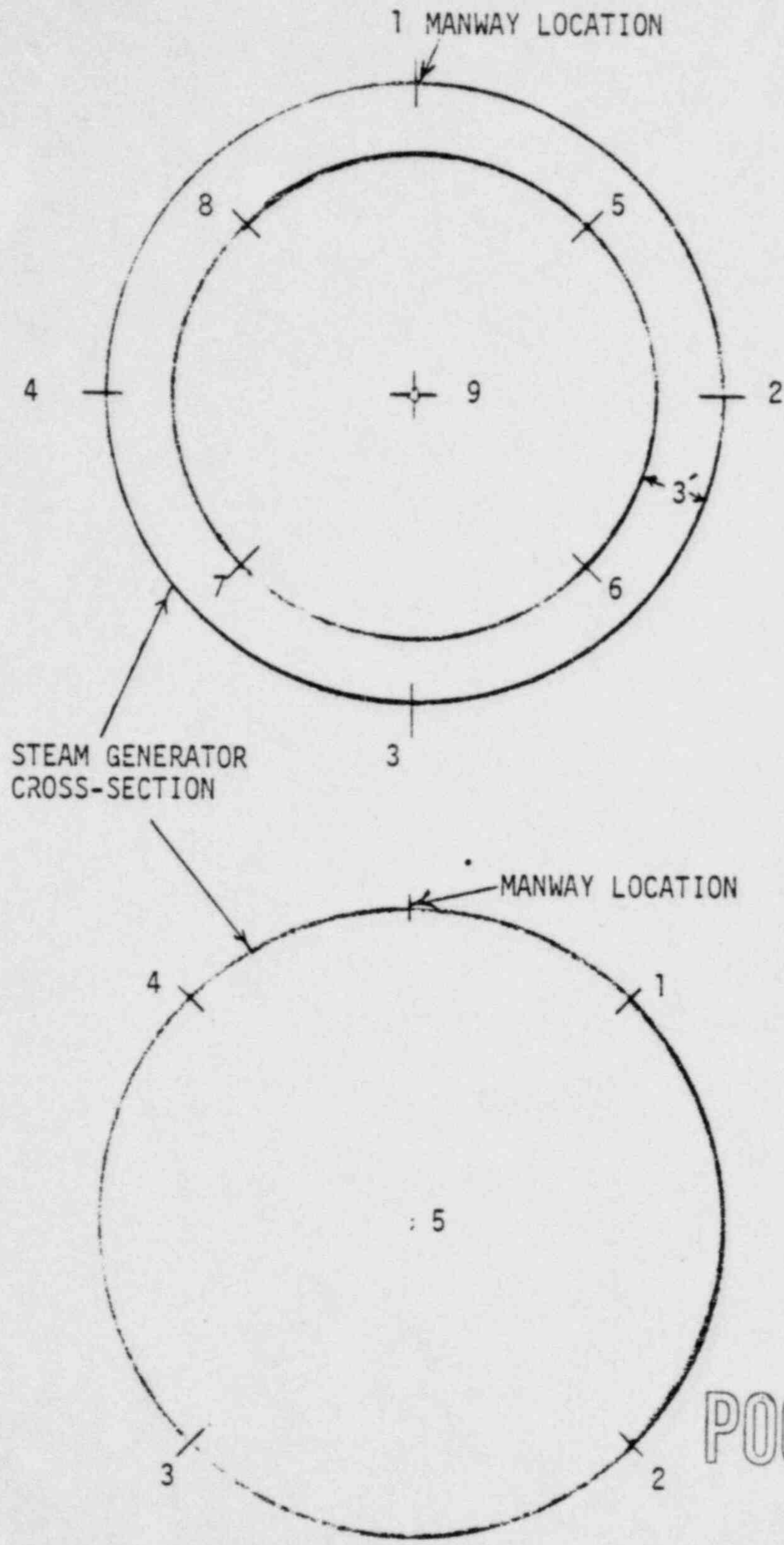
Eddy-current work from the lower head precludes tubesheet shielding for the bottom tubesheet. Portable shielding will be used on the lower dome as needed to reduce exposure, when shielding is compatible with the task to be performed.

Summary

Table 3 summarizes exposures for the total SG repair effort. The estimated total exposure is 126.2 rem. Without shielding, the estimated exposure is 228.5 rem. Without detailed planning, mockup training, and remote operations, the estimated exposure is 600-800 rem.

Based on the analysis presented in this report, it is concluded that the required exposure for the repair effort is as low as reasonably achievable.

TUBE SHEET LOCATIONS



DOME LOCATIONS

FIGURE 1
UNSHIELDED RADIATION LEVEL:

Attachment to Figure 1

Tubesheet Locations

Location Number	Distance Above Tube Sheet (feet)	Radiation Level (R/hr)
1-9	0	4.0
1	2	3.5
2-9	2	3.0

Dome Locations

Location Number	Location on Dome	Radiation Level (R/hr)
1	Sample points located at an elevation just above the manway opening at 45°, 135°, 225°, 315° from the manway. Readings taken directly on the dome surface.	4.0
2-4		3.0
5	Sample point located in the reactor coolant inlet nozzle.	3.0

Table 1

Work Items and Time Required Inside OTSG

ITEM	Time in Head, man-minutes
Install Shielding	
Tubesheet	20
Dome	240
Debur Identified Tubes (includes rework and QA)	2400
Remove Shielding	
Tubesheet	20
Remove lower manway (A and B OTSG)	0
Free Path Check/Eddy Current (both OTSG's)	315
Flush Tubes in B OTSG	10
Install Shielding	
Tubesheet	20
Clear and Eddy-Current Obstructed Tubes	240
Remove Shielding	
Tubesheet	20
Explosive Plug or Stabilize Obstructed or Damaged Tubes	10
Remove Shielding	
Dome	200
Remove J-Leg Screens and Bladders	5
OTSG-A Lower Head Inspection	0

Table 2

Work Items and Exposures Inside B OTSG Upper Head

Work Items	Without Shielding	With Tubesheet Shielding	With Tubesheet and Dome Shielding
	Exposure, Rem		
Install Shielding			
Tubesheet	N/A	1.2	1.2
Dome	N/A	N/A	10
Repair Identified Tubes (includes rework and QA)	140	80	40
Clear and Eddy-Current Obstructed Tubes	14	8	4
Remove Shielding			
Tubesheet	N/A	1.2	0.8
Flush tubes in B OTSG	0.6	N/A	0.4
Install Shielding			
Tubesheet	N/A	N/A*	0.8
Explosive Plug or Stablilize Obstructed or Damaged Tubes	0.6	0.6	0.2
Remove Shielding			
Dome	N/A	N/A	10
Tubesheet	N/A	N/A*	1.2
Total	155.2	91.0	68.6

*Tubesheet shielding is reinstalled to lower exposure required for dome shielding removal. Expsoures for this installation/removal is not needed if dome shielding is not used.

Table 3

OTSG Repair Tasks

Item	Exposure, Rem
B SG Upper Tubesheet Work	66.2
Install/Remove Eddy Current Manipulator, Bottom Head of Both SG's	10.6
SG Lower Head Inspection, Remove J-Leg Screens & Bladders	1.5
Debris Removal, Tube Plugging in Lower Head	17.0
Support Outside SG	14.4
Subtotal	109.7
15% Contingency	16.5
TOTAL	126.2

RB/hw 2140-P