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SUBJECT:
FORWARDING APPLICANT"S RESPONSES TO ITEMS 1. THROUGH 9., ENCL 1 OF NRC LTR DTD 05/02/78, RE CLEANUP OPERATIONS FOR REMOVING DEBRIS FORM THE PRIMARY COOLANT SYSTEM OF SUBJECT FACILITY....W/ATT.

PLANT NAME: CRYSTAL RIVER #3

REVIEWER INITIAL: XJM
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********** DISTRIBUTION OF THIS MATERIAL IS AS FOLLOWS **********

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8003160094 P

Florida Power
CORPORATION

May 16, 1978
File: 3-0-3-a-3

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 No. 3
Docket No. 50-302
Operating License No. DPR-72
Florida Power Corporation

Dear Sir:

Attached are responses to Items 1. through 9. of Part I, Enclosure 1 to your letter of May 2, 1978.

This information is being submitted in accordance with the schedule of activities outlined in our letter to you of May 15, 1978.

Please advise if further discussion on the attached is desired.

Sincerely,

Q. B. DuBois /For
W. P. Stewart
Director, Power Production

luv

A001
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1. Q: Describe cleanup operations for removing debris from the primary coolant system. This should include a description of any grappling, flushing, filtration, and vacuum cleaning techniques to be used. You should also describe which method(s) will be used for each component (e.g. fuel assemblies, reactor internals, steam generators, piping, valves, etc.).

A: Debris removal from the RCS has/will be accomplished by a variety of means. A summary, by component, follows:

Fuel Assemblies: BPRA pins were removed from the guide tubes by mechanical grabbers. Upper and lower end fittings were cleaned with a combination of mechanical grabbers, picks and vacuums. Cleaning was preceeded and followed by a detailed video inspection of all upper and lower end fittings as well as a detailed side view inspection of selected fuel assemblies.

Plenum: Video inspections were conducted and debris removed with a mechanical grabber. A free path check of all control rod guides is planned.

The Reactor Vessel: Vacuumed to remove all debris and video inspected. This inspection included the inlet and outlet piping.

Core Support Assembly: Again video inspections were accompanied with mechanical grabbers, picks and vacuuming. Some debris was simply knocked out through the bottom and will be vacuumed up later.

B-OTSG: Debris was manually removed from the OTSG upper tubesheet and lower head. A visual inspection of the J leg piping showed no debris. All tubes found with debris will be cleaned with a stiff rod and cable. A 100% free path test of all tubes will be followed by eddy current inspections.

A-OTSG: A visual inspection of the upper head revealed no foreign material. Following a free path inspection of 100% of the tubes, the lower head and J leg piping will be visually inspected. Any foreign material will be removed.

2. Q: Describe the cleanup inspection procedures and techniques which will be used. This description should include any methods used to identify the absence of residual debris and the capabilities of the inspection techniques to identify the debris.

A: Cleanup inspection procedures and techniques consist of CR#3 instructions based on approved plant operations and maintenance procedures. They basically give detailed instructions for the removal/disassembly of components within the pressure boundary and debris collection. Inspections for debris have been primarily video using assorted underwater television equipment. Documentation of inspection and cleanup operations is by video tape and independent observation by at least two observers.

All debris observed using video equipment will be removed.* Manual grabbers have removed pieces of debris from 12 feet long down to less than one inch. Vacuuming has removed debris from several inches in length down to debris that appears as specs on the video screen.

3. Q: Describe the damage inspection procedures and techniques used. Identify which components will be inspected for damage, and what criteria will be used to determine the acceptability of any components found damaged.

A: Damage inspection procedures consist of a combination of station approved operation and maintenance procedures and procedures developed by Babcock & Wilcox Company. Components to be inspected include the Reactor Vessel, Core Support Assembly, Plenum, Fuel Assemblies, Control Rod Drive Mechanisms, and Once Through Steam Generators.

Criteria for acceptability is based on the application of each component examined, detailed Engineering evaluation of any damage observed, and inspection of like components in other Babcock - Wilcox NSSS. Documentation of examinations is by video tape and independent observation by at least two observers.

4. Q: Provide the results of the cleanup and inspections discussed above. Itemize the total debris recovered and any debris that is not recovered.

A: Results of the cleanup are analyzed and documented as each operation is completed; including size estimates of the debris recovered. Documentation is by video tape and procedural sign-off. Debris has been removed from the fuel assemblies (upper & lower end fittings and BPRA guide tubes), reactor vessel, core support assembly, plenum and Once Through Steam Generators. Both BPRA spiders and couplings were recovered, one intact in the plenum, one in pieces in OTSG. Of the total

* See attached supplemental Information page.

4. A: Cont'd.

of 403'8" of BPR rod in the two BPRA, total inventory to date is 397'8". Further searches for debris will be free path checks of the reactor plenum, the OTSG A&B tubes, and final cleanup of the Core Support Assembly.*

5. Q: Determine the potential effect(s) that residual poison and metallic fragments will have on plant operations. As a minimum address the following areas:

a. Flow blockage of fuel assemblies. This should include a conservative estimate of channel blockage at the end fittings, grids, and in between grids. You should address the potential for DNB and local cladding hot spots which may cause cladding perforations. The potential for propagation of fuel failures and the means of monitoring and/or mitigating such conditions should also be discussed.

b. The potential for blockage and/or binding of the control rod drive systems due to residual coolant debris. Any procedures planned to mitigate and/or monitor these conditions should be provided.

c. Blockage of the guide tubes which would prevent control rod insertion and safe shutdown operations.

d. Mechanical damage to primary internals due to impacting.

e. Blockage and/or binding of any orifices, valve seats, and vent valves in the primary coolant system.

f. Blockage and/or erosion of steam generator tubes.

g. The effects that the residual debris will have on pumps and any other components with moving parts.

h. Effects on coolant chemistry and crud levels.

A: B&W has evaluated the potential effects of residual poison and metallic fragments and has determined that none of the effects will be detrimental to safe operation. As discussed in question 4 the vast majority of the debris has been removed and any small fragments remaining in the system will soon be flushed to the bottom of the reactor vessel where their effect will be minimal. This position is supported by the previous operation of Arkansas Nuclear Unit One and Oconee Unit Two for several

* See attached supplemental Information page.

5. A: Cont'd.

months with similar size debris in the system with no adverse effects. Considerations was given to the following areas:

a. Flow blockage of fuel assemblies -

The potential effects of residual poison and metallic fragments on DNB are minimal. See response to Question 1 & 2 for cleanup procedure. Any debris left in the system will eventually be carried to the core inlet and become trapped in the lower end fitting or lower end spacer grid. The effect of debris trapped in the lower end fitting has been evaluated using the crossflow codes LYNX1/LYNX2. The results demonstrate that blockage of 20% of the fuel assembly inlet flow area decreases the DNBR by less than 0.1%. A blockage this large is extremely unlikely since it would require several large pieces of debris to be lodged in the same fuel assembly.

It is highly unlikely for debris to work its way into the active fuel region of the core. The largest strip that can fit through a spacer grid would be approximately 0.140" wide. However, if one assumes that a blockage does occur at the spacer grid just below the point of minimum DNBR and that 75% of the flow area in two adjacent channels is blocked, the resulting reduction in DNBR is approximately 5%. This calculation does not consider that turbulence intensities are very high behind the blockage. A study of pressure and flow in a fuel bundle containing blockages conducted by Battelle Pacific Northwest Laboratories measured turbulence intensities five times greater than normal for the area just behind the blockage. This increase in turbulence should offset the loss of flow due to the blockage.

The potential for propagation of fuel failures due to a blockage is extremely remote. The means that is used to postulate the first failure (forcing the coolant from one channel) protects the adjacent channels because more coolant is forced into these channels; thereby, increasing the margin and reducing the possibility of further failures.

The water chemistry is monitored daily and any fuel failure would be detected by this routine inspection.

b. Control rod drive system

B&W inspection of upper plenum cover showed no debris. If coolant debris were ever to reach the Control Rod Drive Mechanisms (CRDM) internals, it would necessarily have to exist in the area between the upper plenum cover and the reactor vessel nozzles. Since no evidence of debris was

found on the plenum cover, the possibility for debris in the mechanism is essentially precluded.

In addition, a Diamond Power Supply Company (DPSC) representative was called to the Crystal River site to inspect the control rod drive leadscrews and closure insert components. The results of this inspection, conducted under the reactor vessel head are: there is no aluminum oxide debris in the CRDM internals.

Further, it was pointed out by DPSC that -

1. Inspection of CRDM components after design life testing have shown that a considerable amount of metallic debris could be present with no detrimental affect on mechanism operation.
2. Inspection of drives which have been ratchet tripped have shown that chips from the leadscrew can be present in the rotor assembly area of the mechanism. Presence of these chips has never prevented a control rod from being tripped or driven into the core.

In summary, based on the above information B&W and DPSC concur that further CRDM inspection is not justifiable and that the CRDM's may continue in normal operation.

c. Possible blockage of control rod guide tubes -

B&W has made an extensive effort to identify and retrieve all the loose pieces from the primary system. Detail description of the efforts made is given in the responses to the question 1 thru 4.

In summary, debris from all the guide tubes of fuel assemblies 3C35 and 3C37 were removed and cleanliness of the guide tubes of these two assemblies were verified by a special probe.

There is a possibility that the small pieces might get into guide tubes and cause some interaction with moving components. However, based on ANO experience, the probability of this occurrence is very small.

d. Possible damage to primary internals -

A detailed inspection of the reactor internals has been carried out and no structural damage detrimental to the function of the reactor internals has been found. In fact the only damage attributable to the loose debris is some minor dings near a large flow hole in the plenum cylinder. This is believed to have been caused by impacting of the LBP spider coupling before the assembly escaped entirely from the

fuel assembly. The fact that no other structural damage was found in the internals, although a significant amount of debris was found on the fuel assembly lower end fittings, the lower internals and the lower head of the reactor vessel, suggests that the parts that are able to pass through the system are too small to cause structural damage.

In conclusion, based on the fact that (1) a detailed inspection of the reactor internals was performed and no detrimental structural damage found, (2) all debris that is found will be removed and, (3) possible remaining debris would be small, no detrimental effects on the function of the reactor internals either present or future are expected from the LBP failures and resulting debris.

e. Effects on RCS valve seats or vent valves -

All vent valves, including the seating surfaces, were visually inspected with a TV camera. This inspection revealed no detrimental structural damage. The only indication of any type was a minor impact mark on one vent valve jack screw, believed to have occurred during removal of the plenum assembly (Plenum assembly was removed from the vessel without the aid of the indexing fixture to facilitate removal of the LBP assembly lodged in the plenum region).

In addition to the detail inspection, the vent valves were exercised and found to operate freely.

In conclusion, based on the results of the visual inspection which revealed no detrimental damage and the fact that the valves moved freely when exercised provides sufficient evidence that the function of the valves have not been impaired.

The possible effects of residual debris on the Pressurizer Safety valves was also considered. Our findings show that any debris particles, which could be drawn from the pressurizer into the safety valves by the suction created when those valves lift, would pass through the valves and into the discharge system without obstructing flow. While marring of the valve seating surfaces could occur and result in leakage after the valve closes, this in no way compromises the safety function of these valves.

f. Effects of Steam Generators -

Video inspections of the OTSG B upper tubesheet have revealed damaged tube ends and tube to tubesheet welds. This damage however is not extensive enough to effect the safe operation of the steam generator.

Erosion of steam generator tubes is not anticipated since all partially attached chips and internally lodged debris will be removed; if the debris cannot be removed the tubes will be plugged. Calculations of the effect of the damage shows insignificant changes to the generator pressure drop and reactor coolant flow characteristics. In order to confirm these conclusions the reactor coolant loop flow signal will be monitored at 40, 75, and 100% power.

As any debris remaining in the system will be in the form of small fragments of little mass, additional damage is not anticipated.

g. Possible effects on Reactor Coolant Pumps -

B&W has reviewed the videotapes of burnable poison rod pieces and spring pieces assumed to have passed through the reactor coolant pumps. None of the pieces shown in these tapes are believed to have had sufficient mass or density to significantly damage the pump impeller on impact. Operational data surrounding the incident is limited. The only data available is verbal, and this data indicates that the pump vibration levels following the incident were comparable to the normal pump vibration levels prior to the incident. In addition, the fact that seal injection was maintained makes it unlikely that any foreign material could have entered the seal areas.

Based on the above, disassembly or inspection of the reactor coolant pump is not warranted. B&W recommends continued operation. Due to the lack of data surrounding the incident, additional conservatism will be added by the following action:

> "Startup and escalation data pertaining to the RCP seals and pump vibration data should be obtained and compared with baseline data for these pumps. This data should be forwarded to B&W for final recommendation and confirmation of our assessment".

h. Effects on coolant chemistry and crud levels -

The effects on coolant chemistry and crud levels are expected to be minimal. The increased boron in solution will be insignificant next to normal soluble boron levels used for plant control (1-2 ppm if all the boron in both BPRA's were dissolved in the coolant).

Suspended debris, including Aluminum Oxide, may have an initial abrasive effect on any crud buildings, but this debris will be removed by the makeup and purification filters subsequent to plant startup.

6. Identify the cause of the BPRA failure addressing possible manufacturing, design, or installation errors. Please include:

   a. A description of the "as found" condition of all BPRA in the reactor. Address any indications of improper seating or wear.

   b. Details of nondestructive inspections of the BPRAs, both damaged and undamaged, and orifice rod assemblies. Address any anomalies found with the holddown latch assemblies.

   c. A description of any destructive examinations that have been performed. Address any metallography that has been completed in the areas of wear.

Response

The cause of the two BPRA separating from their fuel assemblies is still under investigation.

a. Coupling spider assembly of BPRA B-47 was discovered in the steam generator B. The assembly was badly beaten up and was broken up in many pieces. These pieces were collected and sent to B&W's Lynchburg Research Center (LRC) Hot Cell Facilities for visual and dimensional inspection.

   Coupling spider assembly of BPRA B-52 was found in the plenum cylinder with several full and partial length burnable poison rods attached and one locking ball present.

   Many full and part length individual burnable poison rod pieces were found in the guide tubes or upper end fittings of the fuel assemblies from which they came out. A long length of the burnable poison rod was also found wedged into the upper end fitting of an adjacent fuel assembly and a small segment was found lying across the upper end fittings of a nearby fuel assembly.

b. Following defueling at Crystal River 3 (CR-3), all 66 remaining BPRAs were subjected to a lock test and all were found to be locked in their respective fuel assemblies. During removal of the 66 BPRAs, all ball-lock couplings were visually examined; nothing unusual was seen. Nine (9) of the BPRAs were visually examined full-length and 360° around. Nothing unusual was seen. All fuel assembly holddown latch assemblies (68) containing

BPRAs were visually examined 360° around on the inside. Two wear areas were seen on each latch assembly, oriented at 180° to each other.

Three fuel assemblies had wear in the holddown latches which approximated that observed in the holddown latches of fuel assemblies 3C35 and 3C37. While the results of the holddown latch inspections are still being evaluated, preliminary results indicate the wear in the latch assemblies at CR-3 is much higher than the wear observed at Oconee or ANO.

Orifice Rod Assemblies (ORA) at Crystal River 3 were examined as well as the corresponding holddown latches in each fuel assembly. No evidence of wear or any abnormal condition was seen.

Each of the forty ORAs were identified and checked for orientation with respect to the fuel assemblies. The ball latching mechanisms were examined for ball orientation and condition. The holddown latches were examined for evidence of wear, and general visual appearance. Each ORA was reinserted into its corresponding fuel assembly and was verified to be locked in place.

None of the holddown latch assemblies had wear marks, or any features except for two tiny spherical dimples corresponding to the location of the latching balls.

Inspection of ORAs at Oconee, and holddown latches at Oconee and ANO-1 provide additional verification of the observations at CR-3. No evidence of wear or abnormal operation has been seen for any of the ORAs and ORA holddown latches. These results show that ORAs have been used with no failures and no degradation of any kind.

The results of the recent ORA latch mechanism examination firmly supports the current plans of reusing present ORAs. This same ORA design will also be used, as required, to replace BPRAs which are removed. Administrative steps will be taken at Crystal River to assure that the ORAs locking balls are oriented in a direction different than that in which the BPRA locking balls were oriented.

c. Destructive Examination

B&W has not performed any destructive examination on the recovered coupling spider assemblies to date. However, radiographic examination of the coupling spider assemblies was made. Nothing unusual was found which could indicate functional loss of any internal component during the operation.

9. Describe the remedies planned to prevent future occurrence of similar failures.

Response

To avoid future occurrence of similar failure at Crystal River site, BPRAs are replaced by ORAs as stated in response to question 6.b.

Question 7. In your presentation on April 6, 1978, you indicated that the poison rod assembly was lifted out by action of the hydraulic forces within the core. Provide your analysis of this phenomena. The complete analysis should include any simplifying assumptions, conservatisms, and test results used in your evaluation of this phenomena. Describe what provisions are being considered to preclude this condition and how these provisions will effect other plant operations.

Response The lift force on the BPRA was calculated in the following manner: a quarter core LYNX1 model was used to calculate the axial flow and pressure distribution within each fuel assembly containing a BPRA. The formloss coefficients used in this analysis were developed from test data and have been used in all previous Mark-B4 fuel assembly analyses. The lift force was calculated by multiplying the unrecoverable pressure drop times the effective area. The initial calculations predicted a best estimate net lift force of zero to two pounds. A BPRA lift test has recently been completed at Alliance Research Center and the calculational model was adjusted slightly to benchmark the test results. A reanalysis was then performed for the Crystal River 3 BPRA's using the benchmarked model and, as a result, the net predicted uplift force has been revised to three to five pounds.

The lift force on the BPRA is no longer a concern for this plant since all BPRA are being removed from the core.

Question 8 Also during your April 6, 1978 presentation, you indicated that the orifice rod assemblies were lifted by the action of the hydraulic forces. Your basic assumption as to why these assemblies did not experience failure was that they are considerably lighter than the hydraulic forces and therefore are in, essentially, continuous contact with their restraints. This condition was assumed to eliminate, or minimize, the impact (fatique) damage that resulted in failure of the poison rod assemblies. If this is true, provide an analysis on the effects that low flow operations will have on the orifice rod assemblies.

Response The lift force on the ORA was calculated in the same manner as that for the BPRA. The calculated lift force on the ORA is approximately sixty pounds during four pump operation and 35 pounds for three pump operation. The weight of the ORA in water is sixteen pounds. Therefore, for three pump operation the minimum positive lift on the ORA is nineteen pounds. This lift force is fifteen pound higher than the net force on the failed BPRA (under four pump operation) and ten pounds higher than the highest lift force experienced by any BPRA. This margin is sufficient to insure that the ORA's will always be exposed to a positive uplift force during four pump and three pump operation. Furthermore, three pump operation is not the usual operating mode and is used only for limited periods of time. Two pump operation has not been considered because of its limited use.

Supplemental Information

At the present time, all observed debris has been removed from the CSA except for 4 small pieces of BPRA pin. These pieces are located in the lower grid support posts and are estimated to range in length from 4 to 8 inches. If efforts to remove these pieces fail, justification for not removing them will be provided.