



Commonwealth Edison
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Reference (g)

February 25, 1980

Mr. T. A. Ippolito, Chief
Operating Reactors - Branch 3
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Unit 3
Feedwater Nozzle/Sparger and
CRD Return Line Nozzle
Inspection Programs
NRC Docket No. 50-249

- References:
- (a) G.A. Abrell letter to D.L. Ziemann dated September 22, 1976
 - (b) G.A. Abrell letter to D.L. Ziemann dated October 18, 1976
 - (c) M.S. Turbak letter to D.K. Davis dated November 28, 1977
 - (d) M.S. Turbak letter to Mr. Lear dated May 15, 1978
 - (e) General Electric Report, NEDE-21821 dated March 1978, "Boiling Water Reactor Feedwater Nozzle/Sparger Final Report"
 - (f) M.S. Turbak letter to D.K. Davis dated June 23, 1977

Dear Mr. Ippolito:

The purpose of this letter is to provide the feedwater nozzle and the control rod drive (CRD) return nozzle inspection programs to be implemented on Dresden Unit 3 during the present refueling outage. Justification for these programs is provided below, along with the long-term plan for the modification of the feedwater nozzles on this unit.

Feedwater Nozzle Program

I. Feedwater Nozzle/Sparger Inspections

The feedwater nozzle/sparger inspection program for Dresden Unit 3 will consist of the following:

1. Examination of the visible portions of the four spargers using underwater television equipment.

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2. Ultrasonic examination of the inner blend radius and bore of the four nozzles using Procedures NDT-C-24 and NDT-C-25.
3. Ultrasonic examination of the four feedwater nozzle safe ends and safe end welds.
4. Acceptance criteria for the ultrasonic examination shall be identical to that defined in Reference (a), i.e.:
 - a. The calibration piece shall be a duplicate (same material and geometry) of the actual feedwater nozzle and the adjoining section of the vessel wall and associated weld.
 - b. Instrument calibration shall be performed by setting the response of an 8 mm deep notch in the blend radius and bore of the duplicate nozzle to 80% of full screen height (FSH).
 - c. The examination shall be conducted at a sensitivity equal to the calibration sensitivity plus an additional 6 db in accordance with ASME Code, Article I-5112 of Section XI.
 - d. All relevant indications with an amplitude greater than or equal to either 50% of the reference reflector (8 mm notch) or 10% FSH above the clad roll noise level shall be recorded and evaluated. This evaluation shall be in accordance with the methods defined in Reference (b). All evaluations will be made at calibration sensitivity.
 - e. If a relevant indication is evaluated as 80% FSH or more at calibration sensitivity, a dye penetrant examination will be made of the area containing the indication.

II. Justification for the Proposed Feedwater Nozzle/Sparger Inspection Program

On February 2, 1980 Dresden Unit 3 began its third refueling outage following the installation of the interference fit, forged-T, feedwater spargers. The spargers were installed during the 1975 refueling outage occurring on D-3. A complete dye penetrant examination was performed prior to the installation of the new spargers. All indications found were removed by grinding leaving no linear indications. A reexamination of these nozzles was performed in September 1976, during the following refueling outage on that

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unit. An external ultrasonic examination was performed of the inner blend radius and the nozzle bore using CECO. procedures NDC-C-24 and NDT-C-25. As reported in reference (b) no significant indications were found. The unit had accumulated 14 startup/shutdown (SU/DS) cycles.

During the Winter 1978 refueling outage another inspection was performed on the feedwater nozzles. The unit had accumulated 29 SU/SD cycles since the original repair in 1975. The inspection as defined in reference (c) included an ultrasonic examination of the nozzle bore and inner radius of all four nozzles, again using CECO. procedures NDT-C-24 and NDT-C-25. Two reportable indications (Reference (d)) less than 80% full screen height were found in the nozzle bore area of two nozzles. These indications were verified to be the same indications that were found during the fall outage in 1976. It was concluded that these two small indications were cladding discontinuities.

The CECO. ultrasonic testing procedures used for examination of the feedwater nozzles has been demonstrated to be capable of detecting flaws ≥ 4 mm in depth. However, for the purpose of the vessel examination, the procedure requires that an 8 mm notch be used as a calibration reference, which ensures the detection of flaws ≥ 8 mm in depth. The maximum crack, therefore, which might remain after an ultrasonic examination, would be < 8 mm in depth. General Electric has had similar experience with their ultrasonic testing technique as reported in reference (e).

Crack growth curves developed by General Electric and CECO. were formulated assuming leakage flow past the thermal sleeve of the feedwater sparger as was the characteristic of the loose fit spargers. General Electric formulated a curve assuming a generic SU/SD cycle which was later found to be much more severe than the actual operating conditions. This was determined (Reference (f)) while reviewing operating data on Dresden 2 & 3 and Quad-Cities 1 & 2 for the purpose of constructing a plant unique cycle for the CECO. units. It is evident upon comparison of the two curves that the G.E. curve is much more prohibitive towards accumulating SU/SD cycles and continuing unit operation. An 8 mm crack would grow to critical flaw size after 43 SU/SD cycles using the G.E. curve, whereas it would take 68 cycles using the CECO. curve. Comparing the number of SU/SD cycles accumulated on Dresden 3 since the last ultrasonic examination to the empirical crack growth curves, the unit is found to be well within the safe limits of either of the two curves.

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Experience accumulated with the new interference fit sparger, however, has pointed out the conservatism even in the CECo. crack growth curves. Figure 1 contains data accumulated by General Electric on crack depth for up to 75 SU/SD cycles with the interference sparger in use. A curve established using the G.E. generic SU/SD cycle is compared to this actual interference fit data. It can be seen that the worst case of the 10 units with the interference fit sparger has a maximum crack depth of 0.2", with only one other unit having a maximum crack depth of 0.1". The remaining eight units, however, had maximum crack depths that were much smaller or nonexistent. The above data points out the effectiveness of the interference fit sparger in eliminating the leakage flow which is the mechanism initiating the cracking in the feedwater nozzles.

Previous inspections on Dresden Unit 2 and Quad-Cities Unit 2 have confirmed the above trend for plants with interference fit spargers. As reported above, Dresden 2 had cracks less than 1/16" after 33 SU/SD cycles. During the Spring 1978 refueling outage on Quad-Cities Unit 2, a dye penetrant examination was performed on the accessible areas of three nozzles, and the entire bore and inner radius of the fourth nozzle with the sparger removed. The unit had 44 SU/SD cycles since the original repair and sparger installation, and no linear indications were found. Based on the dye penetrant examination, data accumulated by General Electric for the 10 units and on CECo. data for Quad-Cities Unit 2 and Dresden Unit 2, it is our contention that if any cracks exist on Dresden 3, they are no deeper than 0.2" considering that the unit has accumulated 49 SU/SD cycles since the original repair.

Finally, as part of the on-going program to provide a "final fix" solution to the feedwater nozzle cracking problem, CECo. will install the new G.E. double seal/triple thermal sleeve sparger and will remove the clad from the feedwater nozzles on Dresden Unit 3. This work is scheduled to occur during one of the long outages associated with the Mark I containment work scheduled for the Fall of 1981.

In summary, our technical evaluation of the Dresden Unit 3 feedwater nozzle indicates that:

1. All indications on the feedwater nozzle inner radius were removed during the original clad repair and interference fit sparger installation.
2. The ultrasonic examination procedures used will insure that any cracks ≥ 8 mm in depth will be detected.

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3. Conservative crack growth curves still predict that a flaw remaining in the Dresden 3 nozzles subsequent to the previous inspection would be well below the critical flaw size.
4. Feedwater nozzle inspection data from GE and CECO. has proven the effectiveness of the interference fit sparger for providing an end to the effects of the thermal cycling on the feedwater nozzles in that after 75 SU/SD cycles, the deepest crack found to date has been 0.2" (approximately 25 percent of the critical flaw size).
5. Dresden Unit 3 has had 49 SU/SD cycles accumulated since the original repair which is well below the threshold for which significant cracking has been observed on units with interference spargers.

On the basis of these facts, it is judged that the inspection program defined above is adequate. Furthermore, as stated, plans have been made to install the new G.E. double seal-triple thermal sleeve sparger and to remove the nozzle cladding on Dresden 3 during the Mark I, Fall 1981, outage.

Considering the above, plus the fact that a dye penetrant examination of the feedwater nozzles could expend approximately 200 man-rem and 10 critical path days of outage time, we feel that a dye penetrant exam is not warranted this outage. An estimated 18 additional SU/SD cycles, determined from D-3 cycle history, will occur prior to the start of the 1981 refueling outage which is still within the limits of experience with the interference fit sparger. It is the CECO. position, therefore, that the proposed inspection program, even though less stringent than that suggested in NUREG-0312, provides a safe and reliable inspection which would not compromise unit availability.

CRD Return Line Nozzle Program Status

As a result of cracking problems occurring with the CRD return line nozzle in BWR reactor vessels, an inspection of the Dresden Unit 3 nozzle was performed during the Fall 1978 refueling outage. As reported in Reference (d), the inspection consisted of an underwater TV camera examination of the thermal sleeve and an external ultrasonic examination of the inner radius and the wall below the nozzle. During the inspection the nozzle thermal sleeve retainer ring was found cracked and

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the thermal sleeve was subsequently removed. A dye penetrant examination was then performed on the nozzle inner radius and the area below the nozzle. Several linear indications were found all of which were removed by grinding (Reference (d)). The thermal sleeve was not reinstalled.

An UT of CRD return line welds was also performed. A crack was found on the pipe side heat-affected zone of the pipe to safe-end weld. The section of line between the safe-end and the first 90° elbow including the elbow were replaced.

Data on the CRD nozzle cracking problem has indicated that the cracking found in the BWR vessels has been due to thermal fatigue. A metallurgical analysis of the broken pieces of the retainer ring and the cracked pipe confirmed that thermal fatigue was the failure mechanism. (The same test result occurred when the CRD return nozzle thermal sleeve on D-2 was analyzed after having been found cracked).

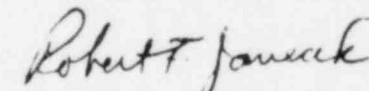
Following the nozzle inspection and repair on Dresden Unit 3, the CRD return line was valved out terminating the 50° - 100°F condensate flow through the return line. Eliminating this cold flow puts an end to the source of thermal cycling which has been determined to be the cracking mechanism.

Based on the above, it is the CECO. position that no further inspection of the CRD return nozzle is warranted. Cracks that were present were removed and the environment that the nozzle will be exposed to will not include the cold condensate flow. However considering the susceptibility of stagnant stainless steel lines to stress corrosion cracking, an augmented inservice inspection will be performed of the stainless steel welds on the reactor vessel side of the inboard valve used for isolation.

Please address any questions you may have concerning this matter to this office.

One (1) signed original and thirty-nine (39) copies of this transmittal are provided for your use.

Very truly yours,



Robert F. Banecek
Nuclear Licensing Administrator
Boiling Water Reactors

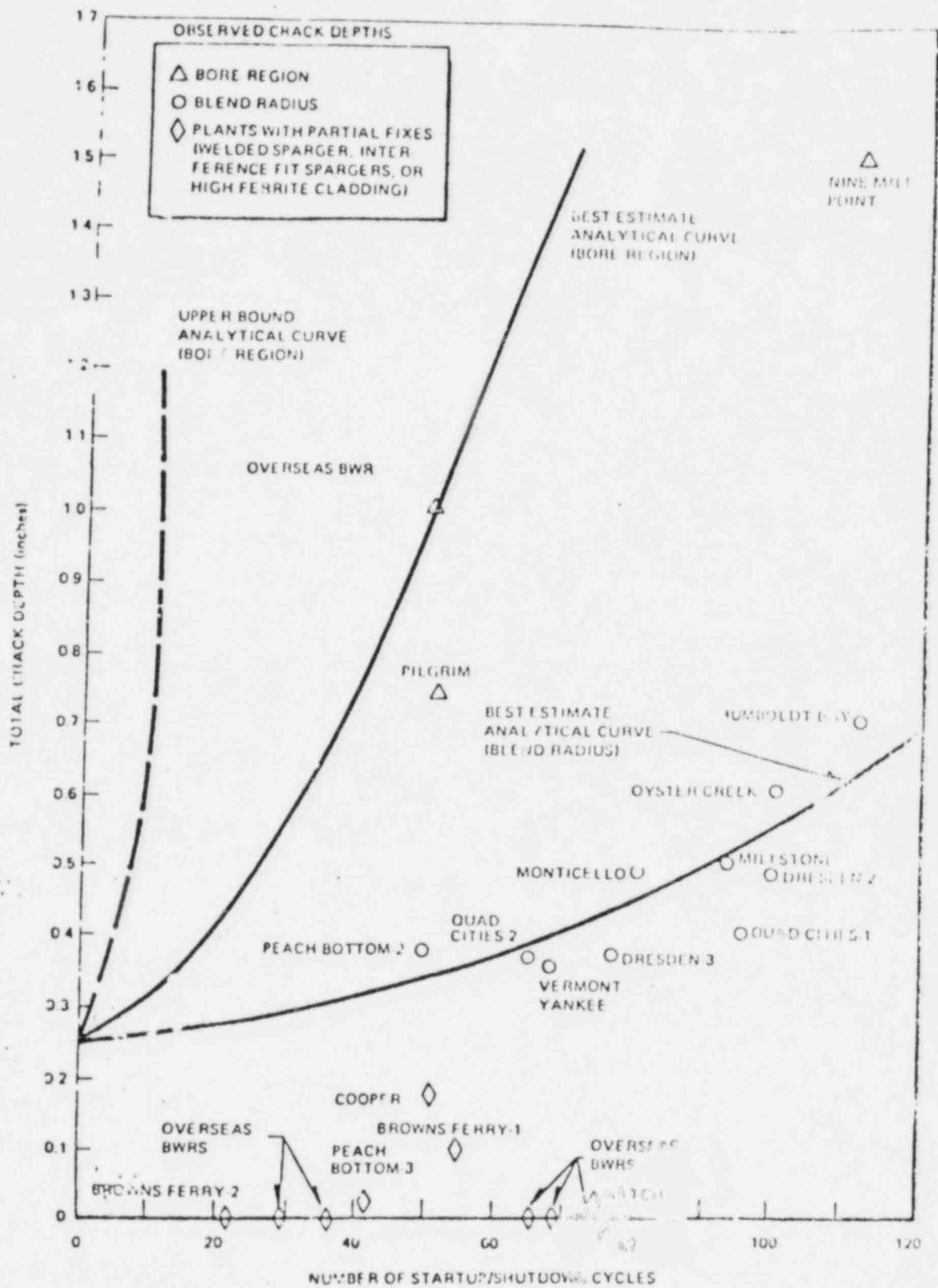


Figure 1



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October 6, 1981

(Ref. h)

Mr. Darrell G. Eisenhut, Director
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Quad Cities Station Unit 2
Implementation of NUREG-0619
Control Rod Drive Return
Line Nozzle Inspections
NRC Docket No. 50-265

- References (a): NUREG-0619, BWR Feedwater
Nozzle and Control Rod Drive
Return Line Nozzle Cracking, -
November 1980.
- (b): R. Janecek letter to D. Eisenhut
dated February 23, 1981.
- (c): T. Novak letter to J. Abel
dated July 20, 1981.

Dear Mr. Eisenhut:

In Reference (b), Commonwealth Edison provided our plans for resolving the Control Rod Drive (CRD) return line nozzle cracking problem (described in NUREG-0619) at Dresden Station Units 2 and 3 and Quad Cities Station Units 1 and 2. In the case of Quad Cities Station Unit 2, we indicated that during the current (Fall, 1981) refueling outage a leak rate test would be performed on the valve which has been used to isolate the CRD return line. This will provide an indication of whether or not there had been leakage of cold water to the nozzle. If the test proves that the valve was leaking, a dye penetrant test (PT) of the return line nozzle as specified in NUREG-0619 would again be performed (a PT examination was performed previously in the Spring 1978, outage). If the valve proves to be leak tight, no further nozzle inspections are deemed necessary because the crack initiating mechanism (cold water) will have been shown to be absent.

In Reference (c), the NRC staff took issue with our position concerning re-inspection of the CRD return line nozzle. The major concern cited was the possibility that crack indications could be discovered in previously inspected and ground-out areas because flaws had been missed by being "battered-over" during the grinding process. It was then concluded that the return line nozzle should be reinspected prior to our implementation of a two valve leak detection modification on the return line.

October 6, 1981

The Commonwealth Edison Company response to this concern and to the other issues raised in Reference (c) will be provided for Dresden Units 2 and 3 and Quad Cities Units 1 and 2 within the time frame requested. However, because Quad Cities Unit 2 is currently shutdown for the Fall 1981, refueling outage, our decision and bases for not re-examining the Quad Cities 2 CRD return nozzle is being provided at this time prior to the Reference (c) requested date.

As committed, the leak rate test on the valve isolating the CRD return line was performed during the current Quad Cities Unit 2 refueling outage on September 24, 1981. At a test pressure of 25 psig, no leakage was found through valve 2-0301-74. The test pressure was mutually agreed upon by the Station and the NRC Senior Resident Inspector, based upon a normal operating differential pressure of approximately 10 psi across the valve, plus an additional margin for conservatism. Because the valve is leak-tight, no further nozzle PT examinations are deemed necessary. Commonwealth Edison strongly believes that not performing a PT of the nozzle will provide no degradation of safety margins, and that our present and already committed programs are both adequate and responsive to the concerns of NUREG-0619. Further justification of this position is provided below.

1. CRD Return Line Nozzle Thermal Sleeve Removal and PT Examination, 1978 Refuel Outage

Diligent efforts were taken in 1978 to assure that the CRD return line nozzle was crack-free so that future PT examinations would not be necessary. The maintenance/modification procedure governing the thermal sleeve removal and PT examination was explicit and thorough concerning the preparation for and conduct of the PT examination, and required Quality Control and Quality Assurance notification prior to proceeding with PT indications grinding. The procedure was prepared by a Maintenance person, and approved by the Master Mechanic, Quality Control Supervisor, Technical Staff Supervisor, Q.A. Inspector and the Assistant Superintendent. The Modification was duly approved and authorized by the Station On-Site Review Committee and the Station Nuclear Engineering Department. A 10 CFR 50.59 Safety Evaluation was performed, and it was concluded that no unreviewed safety questions existed. The PT was performed in accordance with the Commonwealth Edison Company Special Process Procedures Manual. A brief description of the removal and examination follows:

- a. The retainer ring attachment weld was ground out.
- b. The retainer, spring washer, and thermal sleeve were removed. The ring remained in place.

- c. The nozzle area was cleaned with alcohol, and stainless steel wire brushes. The oxide coating was cleaned from the nozzle.
- d. An initial PT was performed, and 4 non-relevant indications were identified. The indications were adjacent to the three nozzle radius thermal sleeve lugs, in the clad portion of the Reactor vessel. The indications were ground out; 2 were to a depth of three-sixteenths of an inch (3/16) and 2 were to a depth of one-sixteenth of an inch (1/16). No grind-outs were deep enough to penetrate into base metal. All grind-outs were finished with a 4-to-1 blend.
- e. A final PT was performed after flapper wheel preparation, and the results were acceptable. All PT was performed by a CONAM Inspection Level II, and verified and accepted by the Station Quality Control Supervisor.

The extensive cleaning and PT preparation measures that were taken were unique to this job, and simple grinding was not done for PT preparation. It is very unlikely that flaws were missed by being "battered-over" during the grinding process. Further assurance that the CRD return nozzle is crack free was provided by subsequent direct visual and underwater TV camera inspection described below.

2. Direct Visual Examination, 1980 Refuel Outage

On February 15, 1980, a direct visual examination of the nozzle revealed no cracking in the nozzle nor in the vessel apron below the nozzle. The inspection was performed by a qualified Level II visual examiner assigned to Station Quality Control Department.

3. Underwater TV Camera Inspection, 1981 Refuel Outage

On September 17, 1981, a Reactor vessel internals inspection was completed on Unit 2. An underwater television camera was used. After focusing, the camera was used to inspect the CRD Return Line Nozzle (N-9). There was not evidence of cracking on the nozzle radius or nozzle apron to an area of about 18 inches below the nozzle. The inspection was performed by a qualified Visual Inspector assigned to the Station Quality Control Department, who has had previous experience in this type of inspection.

4. A nozzle PT in the current refueling outage would require personnel work inside the Unit 2 Reactor vessel at a location where the projected dose rate will be 27 REM per hour. From an ALARA viewpoint and from a personnel safety consideration, this inspection is not justifiable. The risk of potential overexposure also presents itself, should an event such as a power failure to the Reactor Building Crane occur.
5. Lowering the water level in the Reactor vessel with the head removed would be necessary in order for the inspectors to be lowered into the Reactor to do the PT. This would likely be stopping necessary Torus modification and Drywell Hanger outage work on the outage critical path. This is also an undesirable condition from radiological safety and contamination considerations.
6. We plan to close both outboard return line valves (2-0301-74 and 2-0301-94) upon startup of Unit 2 following refueling, and monitor for leakage during subsequent operation. This will provide further redundant assurance that nozzle degradation will not occur.

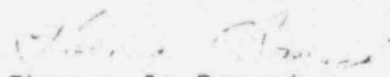
In addition to the above, the CRD return line piping inside the drywell has been UT examined during the current outage per our Reference (b) response. More information concerning this inspection and other issues raised in Reference (c) will be provided for Dresden Unit 2 and 3 and Quad Cities Units 1 and 2 in our next response.

In conclusion, Commonwealth Edison strongly believes that the integrity of the Quad Cities Unit 2 CRD return line nozzle and piping is being maintained, and that inspections performed to date have shown no evidence whatsoever to the contrary. We believe a PT of the nozzle to be unnecessary and potentially hazardous from a radiological and personnel safety standpoint.

Please address any questions you may have concerning this matter to this office.

One (1) signed original and thirty-nine (39) copies of this transmittal are provided for your use.

Very truly yours,



Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors

cc: Region III Inspector - Q.C.
lm