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November 20, 1981

Mr. James G. Keppler, Director
Directorate of Inspection and
Enforcement - Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, IL 60137

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Reports Associated With NUREG 0619,
"BWR Feedwater Nozzle and CRD Return
Line Nozzle Cracking"
NRC Docket Nos. 50-237/249 and 509-254/265

References (a): D. G. Eisenhut letter to Licensees dated November 13, 1980

(b): R. F. Janecek letter to D. G. Eisenhut dated January 22, 1981

(c): R. F. Janecek letter to D. G. Eisenhut dated February 23, 1981 (attached)

(d): T. A. Novak letter to J. S. Abel dated July 20, 1981

(e): T. J. Rausch letter to D. G. Eisenhut dated November 6, 1981 (attached)

(f): M. S. Turbak letter to D. L. Ziemann dated December 5, 1978 (attached)

(g): R. F. Janecek letter to T. A. Ippolito dated February 25, 1981 (attached)

(h): T. J. Rausch letter to D. G. Eisenhut dated October 6, 1981 (attached)

Dear Mr. Keppler:

In response to the Reference (a) request, Commonwealth Edison provided in References (b) and (c) a description of the actions and modifications we are implementing at Dresden Units 2 and 3 and Quad Cities Units 1 and 2 to mitigate the concerns expressed in NUREG 0619 regarding Feedwater Nozzle and Control Rod Drive (CRD)

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return line nozzle cracking. NUREG 0619 requires all licensees to submit various reports concerning these actions and modifications; the purpose of this letter is to document how we have provided and will provide the required information.

There are three categories of reports defined in NUREG 0619:

(1) Section 4.4.3.1, Sparger and Thermal-sleeve-Design Modifications report, including details of on-line leakage monitoring system:

The requested information has already been provided in References (c) and (e), with the exception of the report detailing our on-line leakage monitoring system. To assist your review, we have attached References (c) and (e), and will provide you with a copy of the leakage monitoring system report when it becomes available. (The proprietary version of the report is completed, transmittal is awaiting preparation of the affidavit to withhold from public disclosure, and completion of the non-proprietary version).

Since Dresden Units 2 and 3 and Quad Cities Units 1 and 2 will be using the standard General Electric triple-sleeve sparger design, no justification for this modification is required per section 4.1 of NUREG 0619.

(2) Section 4.4.3.1, Feedwater Nozzle Inspection reports:

Table 2 of NUREG 0619 provides routine inspection intervals as a function of refueling cycles (or startup/shutdown cycles for the routine PT). Commonwealth Edison will provide the requested reports within 6 months of completing an outage at which an inspection was performed.

(3) Section 8.3, reports concerning CRD return line nozzle and inspections:

Commonwealth Edison has already provided (see References (c) and (e), attached) justification for the actions we have taken to resolve the CRD nozzle cracking problem. Since our actions do not involve major modifications, we believe our responses to date are sufficient to satisfy the reporting requirements of this section with the exception of inspection results at Quad Cities Unit 1. The results of our CRD return line nozzle final inspections during the outages in which this line was valved out of service were transmitted previously by References (f), (g), and (h) (all attached) for Dresden 2, Dresden 3 and Quad Cities 2, respectively. The Quad Cities Unit 1 final inspection results will be provided within 6 months after the next refueling outage (now scheduled for Fall 1982).

We are proposing no changes in our normal Inservice Inspection (ISI) Program, however, we recognize the possibility of IGSCC cracking in the 304 stainless steel portions of the isolated return lines. An augmented inspection of the stainless steel portions of the lines will be performed, therefore, in accordance with the recommendations of NUREG-75/067 "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of BWR Plants."

We are also proposing no changes in our CRD maintenance program, as this program already considers detrimental effects of corrosion products in the CRD mechanisms.

Please direct any questions you may have concerning reports associated with NUREG 0619 to this office.

Very truly yours,

Thomas J Rausch

Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors

Attachments

cc: Director of NRC IE
Director, Office of NRR
RIII Inspector - Dresden
RIII Inspector - Quad Cities

2927N

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

February 23, 1981

(Ref. c)

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

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Implementation of NUREG-0619
NRC Docket Nos. 50-237/249
and 50-254/265

References(a): D. G. Eisenhut letter to Licensees dated November 13, 1980.

(b): R. F. Janecek letter to D. G. Eisenhut dated January 22, 1981.

Dear Mr. Eisenhut:

Per our commitment in Reference (b), enclosed for your review is additional information concerning our implementation of NUREG-0619 for Dresden Units 2 and 3 and Quad Cities Units 1 and 2.

Enclosures 1 and 2 provide our implementation plans addressing corrective action for the cracking problems associated with the feedwater nozzle and CRD return nozzle, respectively. As stated, Commonwealth Edison Co. believes that our planned corrective actions adequately address any safety concerns associated with the nozzle cracking problems.

Please address any questions concerning the matter to this office.

One (1) signed original and fifty-nine (59) copies of this transmittal are provided for your use.

Very truly yours,

Robert F. Janecek

Robert F. Janecek
Nuclear Licensing Administrator
Boiling Water Reactors

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cc: RIII Inspector, Dresden
RIII Inspector, Quad Cities

SUBSCRIBED and SWORN to
before me this [illegible] day
of [illegible], 1981

[illegible]

Notary Public

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Enclosure 1
Feedwater Nozzles

Clad Removal and Improved-Design Sparger Installation

CECo's schedule for nozzle clad removal and installation of improved-design feedwater spargers in Dresden Units 2 and 3 and Quad Cities Units 1 and 2 is given in Table 1. The schedule satisfies the NUREG-0619 requirement for completion of this work by June 30, 1983.

The sparger design selected by CECo is the standard GE triple-sleeve sparger which has been approved by the NRC based on review of GE report NEDE-21821-A. Calculations of cummulative nozzle usage factor have been completed for the four CECo plants using conservative triple-sleeve seal leakage assumptions. The results of these calculations, presented in GE report 22A6652 "Feedwater Nozzle Stress Report for Rapid Cycling," show the worst case usage factor to be 0.974 at the nozzle blend radius after completion of the 40 year plant design life (assuming worst case seal degradation and periodic seal refurbishment). Extrapolation of zero leakage results to the end of design life yeilds an end of life usage factor of approximately 0.09 and since our plans include the installation of on-line leakage monitoring systems we expect that the actual end of life nozzle usage factors will be closer to the zero leakage value.

On-Line Leakage Monitoring System Installation

Coincident with the scheduled clad removal and triple sleeve sparger installation, CECo is installing on-line leakage monitoring systems in the four plants. The monitoring systems will detect bypass leakage past the secondary seal by means of thermocouples mounted on the outside surface of the nozzles. Should seal leakage develope the monitoring system will detect the relatively cold feedwater thereby providing early detection, before fatigue damage can occur. This system is identical to the one installed at Monticello where it has been successfully utilized.

Low-Flow Controller

The crack growth analysis results presented in Section 4.7.3 of GE report NEDE-21821-A revealed that reduction of low-flow cycling of feedwater flow will significantly decrease crack growth. In Section 3 of this report the crack growth analysis results are transformed into specific requirements for a low-flow controller that can significantly reduce low-flow feedwater cycling.

While CECo agrees with the analytical results noted above, we do not agree that imposing all of the GE functional requirements described in Section 3.4.4.3 of NEDE-21821-A is the only available method for arriving at a satisfactory low-flow controller. CECo intends to meet the crack growth reduction intent of NEDE-21821-A while retaining the low-flow control systems currently installed. This will be confirmed with plant-specific analyses. Should the analyses indicate that modifications will be necessary to satisfy the crack growth intent they will be completed by June 30, 1983 as required by NUREG-0619.

Reactor Water Cleanup Discharge Reroute

Section 4.7.2 of NEDE-21821-A shows that rerouting the RWCU discharge to each feedwater line results in only a negligable usage factor improvement with a nonleaking sparger (0.05 to 0.06). It only becomes significant when relatively high rates of bypass leakage exist for an extended period of time since the RWCU heating is only significant at low feedwater flow rates (<5%) which occur during approximately 1% of the plants operating time, and since the cyclic amplitude is low at these low flow rates for non-leaking spargers.

The scheduled installation of online leakage monitors at our plants assures early detection of seal leakage thereby minimizing any usage factor improvement associated with this modification. In addition, there has been no field correlation shown to date between nozzle cracking and the presence or lack of RWCU flow. In view of the above, CECo. does not believe that the marginal gain achieved by this modification warrants the high cost required to implement it (approximately $70,000 per station).

In conclusion, the program to be implemented by CECo. is believed to meet the intent of NUREG-0619. Not only are the planned modifications sufficient but also the nozzle cracking problem has added no new safety concerns which have not already been addressed. In our judgement NUREG-0619 does not justify a substantial safety improvement resulting from backfitting these proposed modifications to the RWCU system.

Table 1

SCHEDULE

	Clad Removal & Sparger Replacement	Leakage Monitor Installation	Low Flow Controller Modification (If Required)	RWCU Reroute
resden 2	May '81	May '81	Before 6/30/83	Not Planned
resden 3	Spring '82	Spring '82	Before 6/30/83	Not Planned
uad Cities 1	Fall '82	Fall '82	Before 6/30/83	Not Planned
uad Cities 2	Completed	Completed	Before 6/30/83	Not Planned

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Enclosure 2
Control Rod Drive Return Line Nozzles

CECo. shares the NRC concerns regarding the flow of cold water to the CRD return line nozzle and the subsequent cracking of the nozzle due to thermal fatigue. A review of NUREG-0619 has been performed and a program has been developed for resolving the CRD cracking problem giving careful consideration to the bases for the NRC proposed program.

The CECo. plan will be administered as follows. During the upcoming refueling outages on D-2(Winter 1983), D-3(Winter 1982), QC-2(Fall 1982) a leak rate test will be performed on the valve which has been used to isolate the return line. This will provide an indication of whether there had been leakage of cold water to the nozzle. If the test proves that the valve was leaking a dye penetrant test of the return line nozzle as specified in NUREG-0619 will be performed. If the valve proves to be leak tight no further action is deemed necessary. Subsequent to this test two valves outside containment will be closed for normal operations with a tell-tale drain line located between the two valves being open. The QC-1 nozzle will be dye penetrant tested during its next refueling outage since it has not been previously done. The double isolation will be instituted on the unit following the dye penetrant examination. An augmented inspection of the stainless steel portion of the return lines on all four units will also be performed.

The above program is justified based on reviews of the benefits gained in performing proposed modifications. It has been determined that the CECo. program does not present a safety problem in that the operation of the CRD scram subsystem is not degraded. The following issues were evaluated while developing a program which provides for an acceptable margin of safety:

1) Edison agrees that a dye penetrant examination on the CRD nozzle as required by the NRC is necessary if cold water flow existed during plant operation. However, consider that the Dresden 2, Dresden 3, and Quad Cities 2 lines were valved out of service following the fall, 1977, the fall, 1978, and the spring, 1978 refueling outages, respectively and that during those outages the thermal sleeves were removed and the in-vessel side of the nozzles were dye penetrant examined and repaired as necessary. For these units which have had dye penetrant checks at the time the lines were valved out and for which no leakage can be identified there is no justification for performing additional inspection, especially considering the high doses received in performing such an inspection.

2) In the valved out mode the CRD return line will still be available to provide make up to the reactor vessel. For the highly improbable event in which the line might be

useful access to the reactor building would be available so that an operator could open the isolating valves establishing flow to the vessel. Since the return line will be available, no specific flow test or analysis will be performed.

3) Edison agrees that the potential for IGSC cracking does exist in the 304 stainless steel portions of the isolated return lines. This is due to the susceptibility to cracking of that type of material exposed to stagnant conditions. An augmented inspection of the stainless steel portions of the lines will be performed, therefore, in accordance with the recommendations of NUREG-75/067 "Investigation and Evaluation of Cracking in Austenitic Stainless Steel Piping of BWR Plants". Performing this inspecton is more practical than rerouting the return line and cutting and capping the return nozzle considering the radiation exposure, outage time and rerouting problems (to be discussed later). An estimate of accumulated dose during the above work showed that more total dose would be accumulated in performing the cut, cap and reroute modifications than would be accumulated during augmented inspections performed over the remaining life of the plant.

   Approaching the issue in this manner does not pose any new safety concern and is in itself not a safety issue. Conformance with NUREG-75/067 will preclude any potential safety or reliability concern.

4) It is strongly felt that rerouting the return line only relocates the problem to another area which is just as inaccessible during operation as the containment. Rerouting ignores the primary problem. The problem is that a thermal fatigue prevention device which would alleviate the cracking is not available. Rerouting the CRD return line will provide no additional benefit because thermal fatigue cracking is anticipated at any proposed return nozzle or piping tee. The approach taken by CECo, valving out the return line, eliminates the thermal fatigue cracking.

5) The concerns raised as to the effects of the corrosion products are only valid to a certain degree. The corrosion products do not degrade the operability of the CRD system. This is judged to be the case since regular surveillances are performed on the drives and any irregularities would be noted in advance of normal operation failures. Any rod which does not meet the Technical Specification requirements would be declared inoperable and subsequently left inserted if required. Therefore the surveillances and normal maintenance as performed at this time are deemed sufficient to preclude any CRD failures. Normal operation experience gained on the Dresden and Quad Cities Units with the return line valved out has substantiated the above.

Not only are the surveillance procedures adequate to detect problems but the problems caused by the corrosion products are not as grave as postulated in NUREG-0619. Tests at Dresden have shown that the flow returns to the vessel through the exhaust water header and not through the cooling water lines. Even if the problem were as severe as postulated by the NRC (corrosion products on over piston and under piston sides of the drives) the scram subsystem would not be affected. This is substantiated by the General Electric reivew of corrosion product effects on drives (November 2, 1979 letter from G.G. Sherwood to R.P. Snaider).

6) Finally, the pressure equalizing valves have been determined to be unnecessary. As previously stated the Quad Cities and Dresden Units have been operating in this mode since 1977 for Dresden 2 and Quad Cities 1 and 1978 for Dresden 3 and Quad Cities 2. No problems with system pressures have been experienced in that time. Considering the very low probability of system conditions being such as specified by GE to create the high differential pressure situation plus the Dresden and Quad Cities operating experience no problem is expected. Also, as stated previously, tests have shown that flow goes to the exhaust water header and not for the most part to cooling water lines. Therefore the valves are not necessary to prevent reverse flow to the cooling water header. Lastly, as mentioned in the NUREG the equalizing valves would help prevent continuous reverse flow through the directional control valves. As stated by the NRC this is not of concern based on tests which have been performed by GE and accepted by the NRC.

In conclusion, the program to be pursued by Commonwealth Edison to mitigate the CRD nozzle cracking problems is based on an evaluation of the merits of performing changes and the effects of the CECo proposed program on the safety function of the CRD system. Radiation and economic considerations were evaluated while comparing the gains of either implementing the NRC proposed program or the CECo program. As a result it has been determined that the CECo proposed program addresses all safety issues and results in lower total radiation exposures and will therefore be implemented on the Dresden and Quad Cities Units.

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November 6, 1981

(Ref. e)

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

Subject: Dresden Station Units 2 and 3
Quad Cities Station Units 1 and 2
Implementation of NUREG 0619
Concerning BWR Feedwater and CRD
Return Line Nozzle Cracking
NRC Docket Nos. 50-237/249 and
50-254/265

References (a): T. A. Novak letter to J. S. Abel dated July 20, 1981.

(b): L. O. DelGeorge letter to D. G. Eisenhut dated July 7, 1981.

(c): T. J. Rausch letter to D. G. Eisenhut dated October 6, 1981.

(d): R. F Janceck letter to D. G. Eisenhut dated February 23, 1981.

Dear Mr. Eisenhut:

The following is being submitted in response to the Reference (a) request for additional information or commitments concerning implementation of NUREG 0619 guidance at Dresden Units 2 and 3 and Quad Cities 1 and 2. Clarification or additional information is being provided regarding Commonwealth Edison's position on (1) the need for feedwater low-flow control modifications, (2) the on-line feedwater nozzle leakage monitor leakage system, and (3) the control rod drive (CRD) return line nozzle issues.

Low Flow Control Evaluation

Dresden Units 2 and 3 and Quad Cities Units 1 and 2 have essentially the same feedwater supply and control system configurations. Each unit has three reactor feedwater pumps which are operated independently of each other. Feedwater flow from the three pumps combines in a single 24 inch nominal diameter line, leading to two full flow feedwater regulating valves, installed in parallel in two 14 inch diameter lines. Also, a 20% capacity valve is located in a 4 inch nominal diameter bypass line for use during startup. Low flow control, in the 0% to 20% power range, is accomplished through this bypass valve.

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The bypass valve is an air operated, glove type valve which is controlled by the single element control system illustrated schematically in Figure 1. The control system opens or closes the valve gradually in response to an electric input signal which is proportional to the difference between reactor water level and the desired level set point.

This arrangement of feedwater supply and low flow control system at Dresden Units 2 and 3 and Quad Cities 1 and 2 is similar to that used in the Monticello Nuclear Generating Plant, which has been previously demonstrated through instrumentation and analysis to provide acceptable nozzle thermal cycling in accordance with NUREG 0619 and NRC Generic Letter 81-11 crack growth requirements. Therefore, assuming that the system is operated in a like manner, similarly acceptable thermal cycling is expected. The only significant difference is that at Monticello, the original plug type valve was replaced by a drag valve in order to facilitate use of the automatic level control capability. The globe type valves in place in Dresden and Quad Cities may limit the ability to utilize the automatic control feature at very low power levels (<5%).

Our initial attempts to evaluate the severity of thermal cycling produced by low flow controller operation at Dresden and Quad Cities based on normally accessible plant data have proven inconclusive. Feedwater flow is normally measured through flow elements at the outlet of each of the reactor feedwater pumps, and summed to determine total feedwater flow. This measurement is too coarse to provide meaningful thermal cycling data in the feedwater nozzle region. Therefore, Commonwealth Edison is in the process of implementing a monitoring program similar to that conducted at Monticello, to facilitate our evaluation of low flow controller thermal cycling. This program consists of:

1. Installation of thermocouples on the horizontal length of feedwater pipe in the vicinity of one reactor vessel nozzle.

2. Connecting a recorder to the input demand signal to the feedwater low flow control system.

Data from these two sources will be recorded during initial phases of one of the next convenient startup cycles at one of the four units once the equipment can be installed. The data will be interpreted in terms of amplitude and frequency of nozzle thermal cycling produced during low flow controller operation. The resulting thermal cycling will then be used as input to a fracture mechanics fatigue crack growth analysis to demonstrate whether or not the existing controllers at Dresden and Quad Cities meet the crack growth criteria of NUREG 0619/Generic Letter 81-11. In the event that the above analysis results do not meet the crack growth criteria, modifications will be implemented. These modifications may include change of startup operation procedure and bypass valve type.

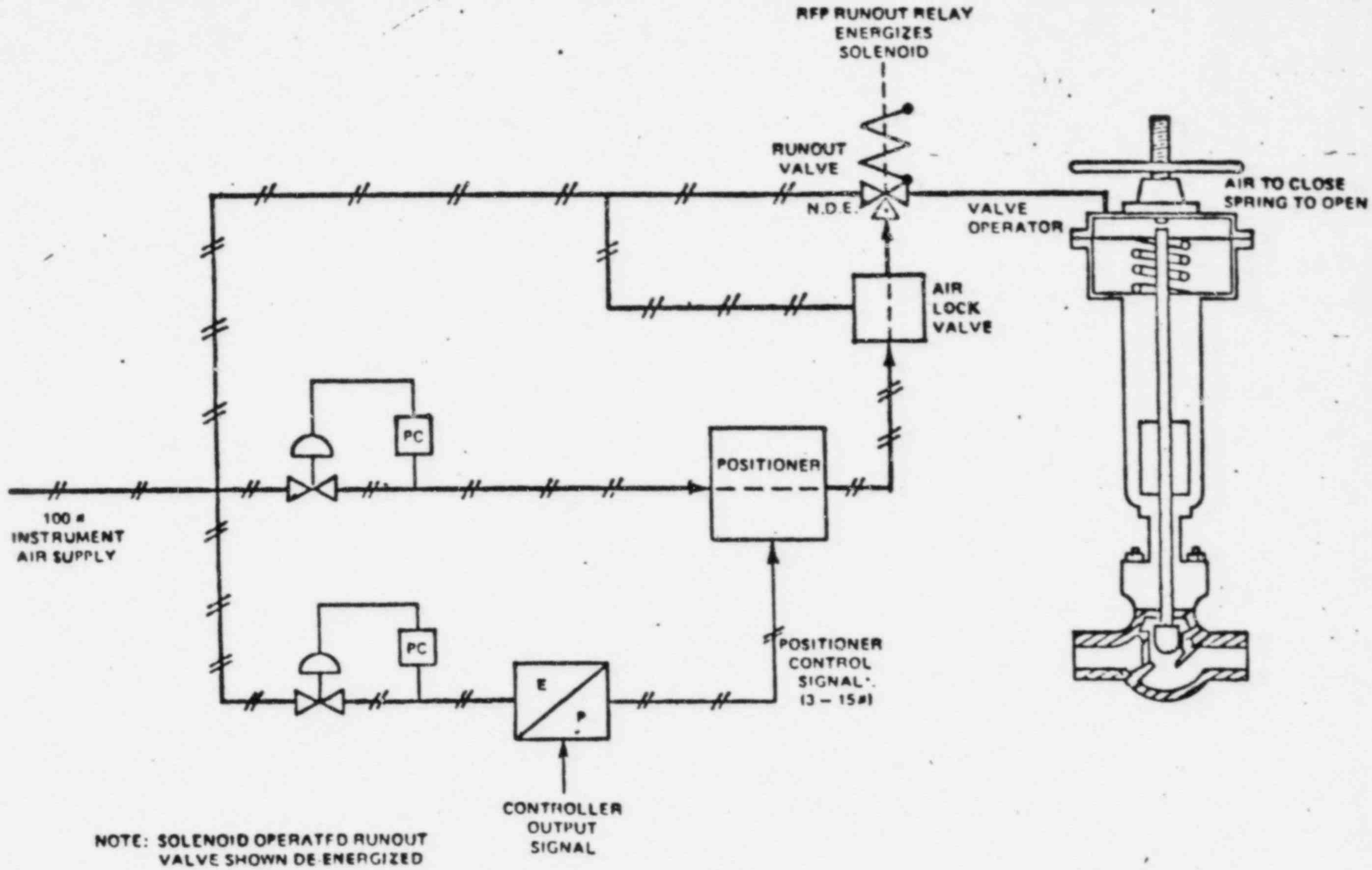

Figure 1. Feedwater Regulating Bypass Valve Control System

On-Line Leakage Monitoring System Installation

On-line leakage monitors are being installed at the Dresden and Quad Cities Stations concurrently with the feedwater nozzle modification program at each unit. Calibration details and technical basis for the systems are documented in the proprietary Licensing Topical Report entitled "NUTECH Feedwater Bypass Leakage Monitoring Systems", ADV-13-002, Rev. 0, (to be provided separately). Leakage monitoring system data from the initial installation of such systems at Dresden Unit 2 and Quad Cities Unit 2 are being collected and analyzed. Commonwealth Edison will submit performance results of the feedwater bypass leakage monitoring system at these units when sufficient field data become available and data evaluation is completed. We feel this information indicates that there is sufficient justification to rely on a temperature monitoring system to assure early seal leak detection. Therefore, Commonwealth Edion still believes that a Reactor Water Cleanup reroute is not justified.

CRD Return Line and Nozzle

Reference (a) identified four areas of concern

1) Length of stainless steel pipe remaining in the drywell and dose needed to inspect the line,

2) Inspection of the nozzle,

3) Type of augmented inspection used, and

4) Need for pressure equalizing valves.

A survey of the sites was conducted which determined that there is approximately 25 feet of stainless steel pipe on both Dresden units and 30 to 35 feet on the Quad Cities units. Recognizing the concern for intergranular stress corrosion cracking (IGSCC), Commonwealth Edison has already embarked on an augmented inspection of these lines. It is planned that at a minimum, 100% of the welds will be inspected every 10 year interval versus the normal 25% coverage. The normal Edison ultrasonic testing procedures which have been audited by NRC personnel are being used. It is estimated that approximately 7R will be expended over the remaining life of the plant to complete this augmented inspection. In comparison roughly 75R per unit would be expended in an effort to replace the existing line with a substitute material. Considering the benefits gained in keeping the CRD return line in place as a source of makeup to the vessel and the dose saved by inspecting the line versus total replacement, it is felt that the augmented inspection as performed by Edison is justified.

The frequency of the augmented inspection should be addressed at this time. The Edison position on the use of NUREG 0313 Rev. 1 is documented in Reference (b). Briefly it states that Edison recognizes IGSCC to be a problem and that augmented inspections are a prudent countermeasure. It is also recognized, however, that with the leak before break scenario and that with the existing leak detection methods available in the plant, IGSCC is not a safety issue. It is an availability issue and therefore any additional inspection (beyond ISI requirements) should be left to the discretion of the owner.

Reference (a) also took issue with the Edison plan for nozzle inspection. This issue was recently addressed in our Reference (c) letter concerning Quad Cities Unit 2. Similarly, at the time the Dresden thermal sleeves were removed, extensive dye penetrant examinations were performed of the nozzle inner radius and the "apron" area below the nozzle. All indications were removed. Since that time, the line has been valved out and no cold flow has occurred. As stated in Reference (d), this will be verified by a valve leak rate check.

The flaws discovered at Dresden did involve more grinding than Quad Cities but it is still felt very unlikely that cracks would have been "buttered-over". Reference (a) noted that this had happened earlier at Dresden. The incident referred to however occurred at an early time in inservice inspectio and repair technology. Since that time, the development and use of sound inspection procedures has for the most part precluded such problems. It should also be noted that such a reinspection would require work in a radiation field of approximately 27 R/hr (Reference (c)) Considering the exstensiveness of the initial inspection, the assurance that leakage has not occurred and the dose involved for inspection, it is felt that the current plan (i.e. inspection if leakage is found) is justified.

Finally Reference (a) states that the NRC does not agree with the Edison position on installing pressure equalizing valves. The corrosion products problem is stated as one reason for requiring the installation of the valves. Contrary to this, it has been determined through surveillances (run on the CRD system when the return line was initially isolated) that the exhaust water does not flow back to the cooling water header.

Reference (a) also indicated that drive performance has been affected at Dresden due to corrosion products. This is true, however, it is strongly felt that the drive wear is being perpetuated by corrosion products which are the result of fuel channel spallation and not due to the carbon steel piping. The Zirconium Oxide which comes from the fuel channels is a very hard and abrasive material more so than corrosion products from carbon steel. Since it is felt that the corrosion products are an in-vessel problem and not one of system operation the installation of pressure equalizing valves is not felt to be necessary. (It should be noted that Commonwealth Edison is pursuing the issue of channel spalling separately.)

The remainder of the potential problems as seen by the NRC which mandate the installation of pressure equalizing valves were addressed in Reference (d), Item II.6. Briefly, it states that based on normal system operations occurring after return line isolation no future problems are anticipated. System pressures have remained normal even with the line isolated at Dresden and Quad Cities Therefore, further action is not deemed necessary.

In conclusion, we believe that our position concerning BWR feedwater nozzle and CRD return line nozzle cracking adequately addresses all safety questions while still considering the cost/benefits of the available options

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

One (1) signed original and fifty nine (59) copies of this transmittal are provided for your use. The proprietary Nutech report will be transmitted under separate cover.

Very truly yours,

Thomas J. Rausch

Thomas J. Rausch
Nuclear Licensing Administrator
Boiling Water Reactors

cc: Region III Inspector - Dresden
Region III Inspector - Quad Cities

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December 5, 1978
(Ref. f)

Mr. D. L. Ziemann, Chief
Operating Reactors - Branch 2
Division of Operating Reactors
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Subject: Dresden Station Unit 2
Feedwater Nozzle/Sparger and
Control Rod Drive Return Line
Nozzle Inspection Programs
NRC Docket No. 50-237

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): G. A. Abrell letter to D. L. Ziemann dated May 17, 1976

(d): M. S. Turbak letter to D. K. Davis dated November 28, 1977

(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

(g): M. S. Turbak letter to D. L. Ziemann dated April 25, 1978

Dear Mr. Ziemann:

This transmittal presents the feedwater nozzle/sparger inspection program that will be implemented on Dresden Unit 2 during the upcoming refueling outage presently scheduled for March 1979. Also included in this letter is the status of the

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control rod drive (CRD) return line nozzle program. Justification for the 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 2 will consist of the following:

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

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

In March 1979, Dresden Unit 2 is scheduled to begin its third refueling outage following the installation of the interference fit, forged-T, feedwater spargers. These spargers were installed during the refueling outage which ended in May 1975. A complete dye penetrant examination of the inner blend radius on the four feedwater nozzles was performed prior to the installation of the new spargers. All indications found were removed by grinding as reported in Reference (c), leaving no linear indicators. A reexamination of these nozzles was performed during the following refueling outage in Spring 1976. The unit had accumulated 19 startup/shutdown (SU/SD) cycles. An external ultrasonic examination was performed of the inner blend radius, using a technique developed in conjunction with Breda Thermomechanica. As reported in Reference (c), no unexplained indications were found. In an effort to validate the ultrasonic testing procedure, a dye penetrant examination was also performed of the accessible areas of the 240^{o} nozzle. Nine cracks were found with the longest two being ground out. Both grindouts were less than 0.070" in depth.

During the Fall 1977 refueling outage on Dresden 2, another inspection was performed on the feedwater nozzles. The unit had accumulated 33 SU/SD cycles since the original repair in 1975. As reported in Reference (d), an ultrasonic examination was performed of the inner radius and bore of all four feedwater nozzles using CECo. Procedures NDT-C-24 and NDT-C-25. One indication was found. It was located on the inner blend radius and was determined to be an existing grind-out. No other reportable indications were found. A dye penetrant examination was also performed on the accessible areas of three of the nozzles, and on the bore and inner radius of the 240° nozzles with its sparger removed. No linear indications were found on the three nozzles. However, nine linear indications were found on the 240° nozzle. Eight of the indications were removed by flapper wheel cleaning, and the ninth was removed by grinding to a depth of less than 1/16". This inspection program, performed in 1977, followed inspection guidelines as set forth in NUREG-0312.

At the present time, Dresden Unit 2 has had 11 SU/SD cycles since the Fall 1977 inspection. Based on previous cycle data for Dresden 2, an estimated 6 additional cycles will occur prior to the March 1979 refueling outage. Performance of the proposed inspection meets the inspection requirements as delineated in NUREG-0312. This document specifies a dye penetrant examination, for vessels which have undergone the original crack removal repair as per the General Electric FDI (performed in 1975 for Dresden 2), be performed at the earlier of (1) every other scheduled refueling outage, or (2) the next scheduled refueling outage occurring after 20, but prior to 40 SU/SD cycles after the last dye penetrant examination.

The CECo. ultrasonic testing procedures used for examination of the feedwater nozzles has been demonstrated to be capable of detecting flaws $\gtrsim$ 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 $\gtrsim$ 8 mm in depth. The maximum crack, therefore, which might remain after an ultrasonic examination, would be $\angle$ 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 2 to the empirical crack growth curves, the unit is found to be well within the safe limits of either of the two curves.

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, and considering Dresden 2 will only have approximately 17 SU/SD cycles at the beginning of the refueling outage, it is our contention that no cracks, if any, are deeper than 0.2".

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 2. At the present time, this work is scheduled to occur during one of the long outages associated with the Mark 1 containment work scheduled for the Fall 1980.

In summary, our technical evaluation of the Dresden Unit 2 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. During the Fall 1977, after an accumulated 33 SU/SD cycles, a dye penetrant inspection in accordance with the guidelines of NUREG-0312 was performed on Dresden Unit 2.

3. The proposed inspection is in accordance with NUREG-0312 in that Dresden Unit 2 will only have an estimated 17 SU/SD cycles prior to the March 1979 outage.

4. The ultrasonic examination procedures used will ensure that any cracks ≥ 8 mm in depth will be detected.

5. Conservative crack growth curves still predict that a flaw remaining in the Dresden 2 nozzles subsequent to the previous inspection would be well below the critical flaw size.

6. 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).

Considering the above facts, plus our plan to install the GE double seal/triple sleeve sparge and to remove nozzle cladding on Dresden Unit 2 in Fall 1980, it is judged that the inspection program is adequate. This inspection program is in accordance with that specified in NUREG-0312 and will provide a safe and reliable inspection which will not compromise unit availability.

<u>CRD Return Line Nozzle Program Status</u>

As a result of cracking problems occurring with the CRD return line nozzle in BWR reactor vessels, an inspection of the Dresden Unit 2 nozzle was performed during the Fall 1977 refueling outage. As reported in Reference (g), 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. The nozzle thermal sleeve was found cracked and was subsequently removed. A dye penetrant examination was then performed on the inner blend radius and bore of the nozzle. Several indications were found in one of the thermal sleeve spacer pads and in the retainer ring pad. These were removed by grinding and were found not to penetrate vessel cladding. The thermal sleeve was not reinstalled.

Data on the CRD nozzle cracking has indicated that the cracking found in the BWR vessels has been due to thermal fatigue. A metallurgical analysis of the thermal sleeve confirmed that thermal fatigue was the failure mechanism.

Following the nozzle inspection and repair on Dresden Unit 2, 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 the 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 line nozzle is warranted. Cracks that were present were removed, and the environment 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.

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

Very truly yours,

L. O. DelGeorge

for/ M. S. Turbak
Nuclear Licensing Administrator
Boiling Water Reactors

attachment

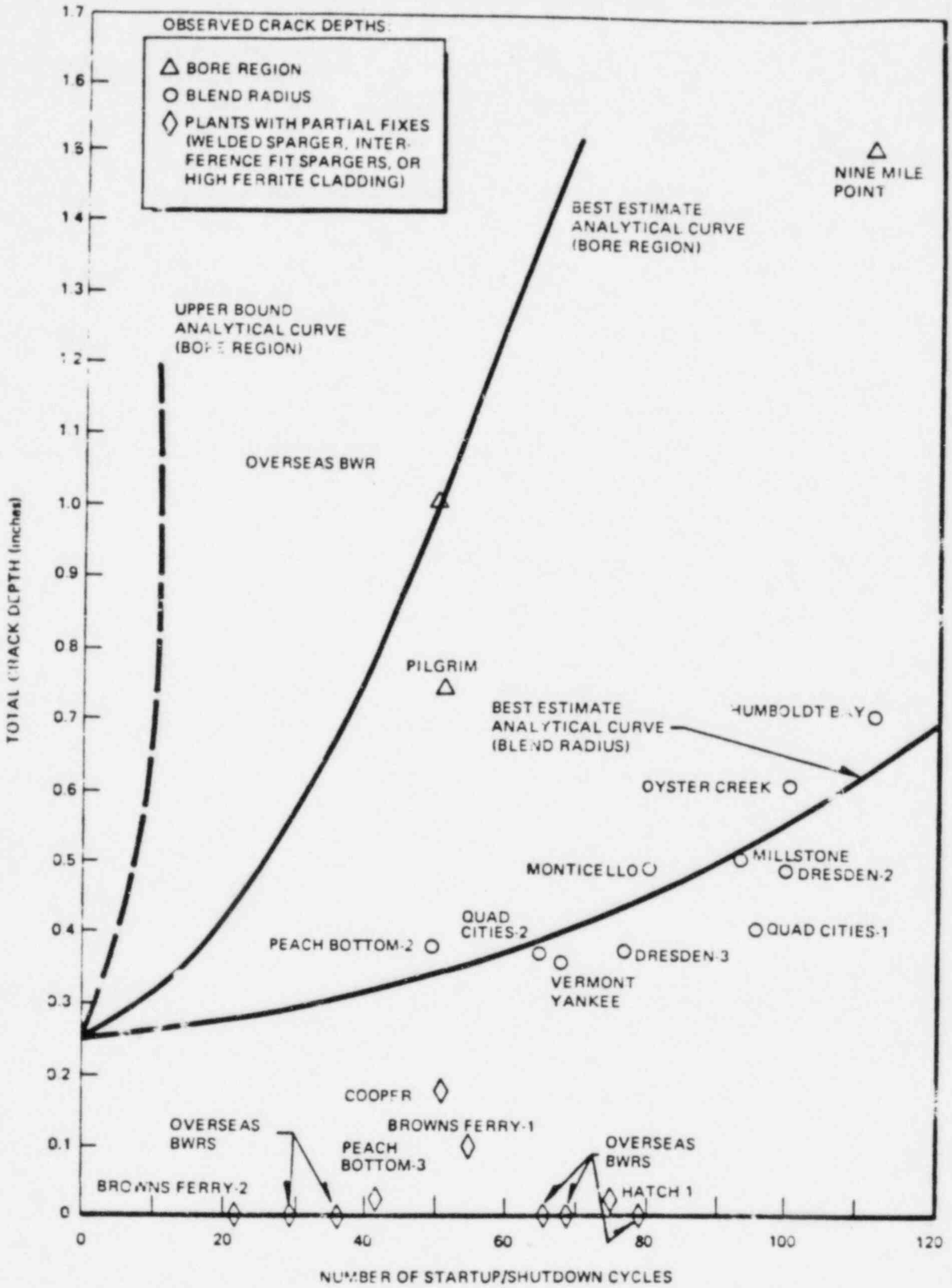
OBSERVED CRACK DEPTHS:
BORE REGION
BLEND RADIUS
PLANTS WITH PARTIAL FIXES (WELDED SPARGER, INTERFERENCE FIT SPARGERS, OR HIGH FERRITE CLADDING)
NINE MILE POINT
BEST ESTIMATE ANALYTICAL CURVE (BORE REGION)
UPPER BOUND ANALYTICAL CURVE (BORE REGION)
OVERSEAS BWR
PILGRIM
BEST ESTIMATE ANALYTICAL CURVE (BLEND RADIUS)
HUMBOLDT BAY
OYSTER CREEK
MILLSTONE
MONTICELLO
DRESDEN-2
QUAD CITIES-2
QUAD CITIES-1
PEACH BOTTOM-2
DRESDEN-3
VERMONT YANKEE
COOPER
OVERSEAS BWRS
BROWNS FERRY-1
PEACH BOTTOM-3
OVERSEAS BWRS
HATCH 1
BROWNS FERRY-2
TOTAL CRACK DEPTH (inches)
NUMBER OF STARTUP/SHUTDOWN CYCLES
0
0.1
0.2
0.3
0.4
0.5
0.6
0.7
0.8
0.9
1.0
1.1
1.2
1.3
1.4
1.5
1.6
1.7
0
20
40
60
80
100
120

Figure 1