NUREG-0312

INTERIM TECHNICAL REPORT ON BWR FEEDWATER AND CONTROL ROD DRIVE RETURN LINE NOZZLE CRACKING

1813 334

Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission

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Division of Operating Reactors
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

TABLE OF CONTENTS

PART I

FEEDWATER NOZZLE CRACKING

1813 338

1.0 INTRODUCTION

Recent operational experience has revealed significant degradation (cracking) of the inner surface of BWR feedwater nozzles. While the problems of feedwater nozzle cracking have not been completely resolved, the NRC staff believes that sufficient information is available from various sources to warrant a discussion of the elements of the problem and of potential remedial measures, both long and short term, which are under consideration. The purpose of this report is to summarize this problem area and to present an interim staff position regarding this generic issue.

It should, of course, be recognized that information on this subject is continually being generated, hence the staff position and comments are subject to further modification. In this regard, we have requested licensees and the General Electric Company to inform us promptly of any information regarding this subject that results from on-going programs and related experience.

The following discussion is based primarily on information supplied to the NRC by the General Electric Company, and on the staff's case-by-case reviews of feedwater nozzle inspection results from a number of operating BWR facilities. The staff has prepared, as part of this report, an interim inservice inspection position to ensure an appropriate conservative treatment of this potential problem at operating facilities until a long term solution is developed.

2.0 STATEMENT OF THE PROBLEM

The feedwater nozzles of essentially all operating BWR's[1/] which have been inspected to date have been found to have blend radius cracks, some of which propagated through the cladding into the base metal. In several reactors, similar cracks were found in the nozzle bore.

1/ Cracks have been observed at all operating BWR's inspected to date with the exception of Browns Ferry 2, which has had less than one year of operation.

The deepest cracks found to date were in the nozzle bore and were of a total depth of about one and one-half inches. Examples of such cracking are given in Attachment 1. Analyses by the NRC staff, which are in agreement with those done by GE and field data from operating BWRs, indicate that the initial crack growth rate is high up to crack depths of about 1/4 to 1/2 in. Further growth is slow but would accelerate with increasing depth. Eventually, the cracks present a repair problem if, in removing them by grinding, the ASME Code limits on nozzle reinforcement were exceeded. The crack depth equated with the reinforcement limit will depend on the details of nozzle dimensions (see NB-3330 in Sect. III, ASME Code).

Feedwater nozzle cracks are of concern to the NRC staff because: (1) reactor pressure vessel integrity is considered extremely important to safety, (2) there are uncertainties about the rate at which the cracks are growing, (3) current nozzle repair procedures require that cracks be ground out thus removing metal from a relatively high stressed region of the reactor vessel, and (4) considerable radiation exposure is received by personnel performing inspections of the nozzle region and repairing cracks in the nozzles. Although such cracking of the pressure vessel nozzles is important to safety, the NRC staff believes that cracking that has penetrated the vessel cladding will grow at a slow enough rate such that the cracking does not pose a critical safety concern today that warrants immediate action. Rather, the staff believes that sufficient time is available, due to the conservative design of the reactor pressure vessel, to permit continued operation of the affected facilities while studies on these events continue on an expedited schedule.

3.0 CAUSE OF THE PROBLEM

The NRC staff is in general agreement with GE as to the mechanisms responsible for crack initiation and growth. Crack initiation is believed to be the result of high cycle thermal fatigue caused by fluctuations in water temperature within the vessel in the sparger-nozzle region during periods of low feedwater temperature when the flow may also be unsteady and perhaps intermittent. Attachment 2 is a trace of such temperature fluctuations as they were measured in a mock-up of a typical feedwater nozzle. Once initiated, the cracks are believed to be driven deeper by the larger, relatively low frequency, startup/shutdown pressure and thermal cycles. The latter result from significant changes in feedwater temperature during flood-up of the reactor vessel and when feedwater heaters are put into, or taken out of, service. During normal power operation, the plant feedwater heaters maintain the feedwater temperature at about 180°F below the reactor water temperature. At low power, when the feedwater heaters are not in service, the temperature differential can be 400°F or more. We believe that the basic cause of the thermal fatigue cracking problem is this relatively large temperature differential between cold incoming feedwater and the hot reactor vessel water during low power and flood-up operations.

4.0 ULTRASONIC INSPECTIONS OF FEEDWATER NOZZLE INNER RADIUS

A number of ultrasonic (UT) examination techniques presently are used to inspect the feedwater nozzle inner radius from outside the vessel. While these inspection methods are useful, their current reliability is limited due to the unique character and location of the thermal fatigue cracks.

Based on our review of the available field examination results, we conclude that the UT methods, when applied to nozzle geometry, have not demonstrated a level of reliability that would allow UT to be used as a sole basis for a decision to permit continued reactor vessel operation. To improve confidence in this method, we encourage the continued development and use of UT techniques for the feedwater nozzle inner radius examinations. Should future developments and examination results demonstrate the UT techniques reliably and consistently detect thermal fatigue cracks in the nozzle region, these techniques could then be used as a basis for modifying the Interim Criteria discussed below.

5.0 INTERIM PROCEDURAL MEASURES FOR OPERATING REACTORS

Because of the current incomplete status of studies and design efforts to resolve the nozzle cracking issue and because hardware changes and other long term remedial measures will require considerable time to implement at operating facilities, certain interim revisions in operational practice are desirable.

In general, the NRC staff has concluded that BWR facility operators should monitor feedwater temperature and flow during low power operation. In addition, operating procedures should be revised to minimize rapid changes in feedwater flow and/or temperature, to minimize the duration of cold feedwater injection, to avoid conditions that may lead to inadvertent or unnecessary high pressure coolant injection (HPCI) system actuation, and to avoid the introduction of cold water from the reactor cleanup system. Reactor operators should attempt to limit the temperature differential between water entering the feedwater nozzles and the reactor vessel water to no greater than the normal differential at full power. They should also avoid feedwater temperature transients to the extent practicable. It has been demonstrated that by carefully bringing feedwater heaters into service, the magnitude of feedwater temperature transients can be significantly reduced. While these steps are not expected to eliminate the nozzle cracking problem, we believe that they should help to minimize the extent of cracking until permanent changes are made.

The NRC staff has held numerous discussions with GE and with licensees on a case-by-case basis to consider this issue. In general, licensees have been implementing the BWR Feedwater Nozzle Inspection interim criteria (Section 6, below) for more than a year and have adopted many of the staff's recommendations for minimizing the cold water flow through feedwater nozzles. There are extensive developmental efforts to perfect ultrasonic nondestructive examination procedures underway at GE and several utility groups. Cladding removal is underway in three reactor vessels and other utilities are planning similar action. In summary, the staff believes that the feedwater nozzle cracking problem is being mitigated satisfactorily on an interim basis while long term remedies are being developed. Attachment 3 is a table summarizing feedwater nozzle cracking experience to date.

6.0 INTERIM CRITERIA FOR BWR FEEDWATER NOZZLE INSPECTION

6.1 Introduction

Analyses performed by the NRC staff and by GE indicate that thermal fatigue cracks in feedwater nozzles can initiate and grow rather rapidly to depths of 1/4 to 1/2 inches primarily due to water temperature fluctuations in the vicinity of the nozzle during operations with unheated feedwater flow. This growth can be experienced within the first fuel cycle of operation. Further growth of these cracks is at a slower calculated rate. The results of feedwater nozzle inspections reported to date appear to confirm the analytical predictions. There are uncertainties associated with the analyses, operations and inspections however, as indicated by some discrepancies between the inspection results from one facility to another or even between nozzles of the same reactor. The objective of this inspection program is to ensure that no cracks grow to a depth where they become safety significant or where the repair procedures to eliminate them would pose a problem.

Although a detailed review of the results of feedwater nozzle inspection and repair at the many facilities (see Attachment 3 for a summary), domestic and foreign, which have taken such actions would be beyond the scope of this document, the activities at two plants should be mentioned.

The Niagra Mohawk Power Company, Nine Mile Point (NMP), facility and the Jersey Central Power & Light Company, Oyster Creek facility, were among the first BWR plants to go into operation (both in 1969). As such, they have accumulated a relatively large number of startup/shutdown cycles[2/] (both about 100) and have "sister" reactor pressure vessels. Using a machine designed for the specific task, both utilities will remove the feedwater nozzle stainless steel cladding during the 1977 refueling outages. As of mid-June, the four nozzles at Nine Mile Point (NMP) were finished and the job at Oyster Creek was underway. After machining at NMP, penetrant testing revealed 5 crack-like indications on one nozzle, one indication on an adjacent nozzle and none on the other two. The depth of local grindout required to remove the single indication and the deepest of the other 5 was about 1-1/2 inches; the length and width of the oval (region of contour grinding) were about 9-1/2 in. and 4 in., respectively. The four nozzles at the Oyster Creek facility were penetrant tested before machining and although the degree of cracking varies from nozzle to nozzle, the longest were of the same magnitude as those at NMP. We therefore expect the final grindout to be approximately the same as for NMP. The removal of such a large amount of material from the pressure vessel makes an analysis of the reconfiguration, relative to the ASME Boiler and Pressure Vessel Code, mandatory.

For the NMP reactor, the applicable Code for nozzle reinforcement is Section III, article NB-3330. Detailed calculations have shown that excess reinforcement remained at the deepest local grindout, using a conservative method of calculation.

2/ The change in reactor thermal power from nominally zero to operational level and return to zero; see Sect. 6.2, below.

Section III of the Code also requires a fatigue evaluation which was performed in a conservative manner for the deepest grindout showing that there was adequate margin between the resulting calculated fatigue life (to crack initiation) and the vessel design life of 40 calender years. Based on past crack propagation analyses, an undetected crack remaining in the nozzle after rework will not grow significantly during the following fuel cycle. The adequacy of any other such clad removal machining operations will be evaluated similarly to that for NMP.

The NRC staff has considered a number of alternative approaches for monitoring and limiting the growth of feedwater nozzle cracks in operating BWR's during the interim period while a long term solution is being developed. On November 19, 1976, the General Electric Company issued a Feedwater Nozzle Interim Examination Recommendation (FNIER) as Service Information Letter (SIL) No. 207. This document, in effect, measures service time in terms of the number of startup-shutdown cycles. The staff also examined service time as measured in terms of the number of hours that the reactor water is hot while the incoming feedwater is cold, that is, the duration of unheated feedwater flow. We believe that this is the mode of operation during which cracks are initiated and grow to the order of 1/4 in. in depth, hence, the number of cold feedwater hours is an appropriate parameter for at least the first fuel cycle. Subsequent crack growth appears to be more closely related to the thermal and pressure stress cycles associated with startup and shutdown. Although there is not necessarily a direct relationship between cold feedwater hours and startup/shutdown cycles, these approaches are not entirely inconsistent and, in view of the statistical nature of the inspection results to date, we find the approach in SIL No. 207 to be generally acceptable.

Until other procedures have been qualified to the satisfaction of the NRC, we have concluded that BWR feedwater nozzles should be inspected in accordance with the program set forth in Section 6.3 below. The NRC staff approach generally agrees with the inspection frequencies and actions recommended by GE; however, in some cases the staff's conclusions are based on the number of hours of cold feedwater flow as well as on the number of startup/shutdown cycles. Therefore, licensees should keep adequate records of these parameters. In addition, licensees should provide temporary instrumentation to monitor detailed feedwater temperature and flow during at least several startup and shutdowns. The NRC recommends that ultrasonic (UT) procedures be used in conjunction with dye penetrant (PT) testing to the extent practical to expedite their development.

6.2 Background Information

The recommended plan, described below, is applicable to all BWRs with feedwater nozzles that do not have the thermal sleeve welded to the nozzle safe end and for plants that went critical after 1968. Other BWRs will be evaluated on a case-by-case basis by the NRC staff.

In the context of this document a startup/shutdown cycle is defined as a reactor thermal power increase from nominally zero and subsequent return to zero which produces both pressure and temperature changes and involves the addition of any amount of cold feedwater through the feedwater nozzles. UT refers to ultrasonic inspection performed from outside the reactor vessel. PT refers to liquid penetrant inspection performed from inside the reactor vessel.

6.3 Inspection Program

The following is an outline of the procedures for performing the required inspection program.

1. At each scheduled refueling outage, perform an external UT examination of all feedwater nozzle safe ends, bores, and inside blend radii. If indications are found in the safe end, evaluate per ASME Section XI. If reportable indications are found in the nozzle bore or the nozzle corner, proceed with the sparger removal PT inspection and repair called for in item 3 below.

2. Determine from plant records the number of startup/shutdown cycles for the reactor.

3. The first feedwater nozzle inspection should be performed after about 50 startup/shutdown cycles but prior to 70 cycles. The following should be performed:

   (a) externally examine by UT all feedwater nozzle blend radii, nozzle bores, and safe ends; (b) remove a sparger from one nozzle, flapper wheel grind and PT examine both the nozzle for the removed sparger and accessible portions of the other nozzles. If any cracks are detected, remove all spargers and completely examine all nozzles. Remove all nozzle cracks.

4. Fcr those plants where the feedwater nozzles have been PT examined but the flapper wheel cleaning or removal of all detected cracks was not performed, the nozzles should be reinspected as per item 3 above at the next scheduled refueling outage.

5. For those plants where the feedwater nozzles were PT examined per GE Field Disposition Instruction (FDI) recommendations, and all detected cracks were removed, subsequent PT examinations of the nozzles should be performed at the earlier of: (a) every other scheduled refueling outage, or (b) at the scheduled refueling outage after 20 but prior to 40 startup/shutdown cycles after the last PT examination.

6. If the feedwater spargers have forged tees, the PT examination may consist of flapper wheel cleaning and PT examination of the accessible portions of the nozzle. If the feedwater spargers do not have forged tees, a sparger should be removed from one nozzle. Then that nozzle and the accessible portions of the other feedwater nozzles should be flapper wheel cleaned and PT examined. If any nozzle cracks are found, remove all spargers, clean, examine and repair the nozzles.

In addition to the above procedure, the depth of cracks that penetrate into the base metal or that are in excess of 1/4 inch deep should be measured and recorded and a record should be made of the circumferential and axial position of each crack.

The sum of the total depths of all cracks that penetrated into base metal or exceeded 1/4 inch should be determined as well as the clad depth of several locations. If any crack exceeds 3/4 inches total depth or if any crack penetrates deeper than 1/2 inches into base metal, a safety analysis report which includes a discussion of the proposed repair procedure should be submitted to the NRC for review and approval prior to further action.

Inspection results should be communicated (orally or in writing) to the NRC as soon as practicable after results are obtained. Such an approach will better insure that the NRC staff can respond to licensees' requests in a timely manner. Spargers and thermal sleeves should not be re-inserted in the vessel until the results of the inspection have been discussed with the NRC. In addition to prompt transmittal of inspection results, licensees are requested to coordinate their planning for feedwater nozzle inspections with the NRC as early as practicable to minimize downtime and potential questions as to NRC requirements.

In determining the inspection frequency of a specific facility, the NRC staff will also consider any remedial measures a licensee may previously have taken to mitigate the feedwater nozzle cracking problem such as:

. Measures to reduce significantly the duration of periods of low feedwater temperatures;
. Significant reduction of maximum ΔT between reactor water and incoming feedwater by system modifications and/or operating procedure changes;
. Appropriate and approved design modifications to the nozzle-thermal sleeve-sparger region that shield the nozzle bore and blend radius surfaces from significant coolant temperature transients;
. Repairs that result in a more highly fatigue resistant nozzle surface condition, as demonstrated by analysis and testing; and/or
. Other remedial measure that minimize thermal fatigue cracking.

7.0 RECOMMENDED ACTIONS

7.1 Feedwater Nozzle Inspection

In view of the status of the overall BWR feedwater nozzle cracking experiences to date, we have concluded that it is appropriate and necessary to thoroughly inspect such nozzles when affected nuclear facilities are shut down for refueling. Accordingly, licensees should submit their proposed inspection plans to the NRC, including the number of nozzles to be inspected, inspection technique(s) and acceptance criteria to be utilized, methods of nozzle surface cleaning, planned actions if UT or PT indications are found, and a synopsis of startup/shutdown cycles, hours of unheated feedwater flow, and operating transients/conditions experienced which are germane to the nozzle cracking phenomenon such as inadvertent HPCI initiation or unstable feedwater flow control. These should be submitted at least 90 days prior to the projected start of the reactor refueling outage.

7.2 Occupational Radiation Exposure

10 CFR Part 20.1(c) states that licensees should make every reasonable effort to keep radiation exposures "as low as is reasonably achievable" (ALARA). The inspection and repair of reactor vessel nozzle cracks has a potential for significant occupational radiation exposures because of the high radiation levels in the work areas and relatively long stay time required to perform the necessary work.

Consequently, licensees are requested to provide a description of the plans and procedures that will be implemented to keep radiation exposures ALARA during proposed nozzle-related work. The description should address the following areas:

(1) Training programs, including use of mock-ups, which will be used before beginning the actual repair to ensure minimum stay times for completion of the job.

(2) Special tools which will minimize personnel stay times.

(3) Shielding used to reduce radiation levels.

(4) Use of decontamination (such as hydrolasing) to reduce radiation levels.

Upon completion of the nozzle related work, licensees are requested to provide a description of the experience so that other licensees can benefit from it in planning their ALARA programs. The description should include the following:

(1) Dose rate information in critical areas before and after decontamination; shielding installations and their efficacy.

(2) Numbers of workers involved in the entire operation.

(3) Total man-rem exposure for the operation and man-rem breakdown by specific phases and by occupation, if available.

8.0 LONG TERM RESOLUTION OF THE PROBLEM

The above staff interim criteria are primarily meant to establish a basis for continued plant operation during the time required to develop and implement long term solutions to this problem. As further information becomes available the staff will continue its review in this area and will issue final criteria at the appropriate time.

The ultimate remedy which will preclude feedwater nozzle cracking may require a combination of individual measures to eliminate the severe thermal transients or make the nozzle less vulnerable to them. Such measures could include reduction of the feedwater to reactor water temperature differential during low power operation, an improved thermal sleeve-sparger design to reduce bypass flow which exposes the nozzle surface to fluctuating water temperatures, and removal of clad from the nozzle surface which is believed to provide a surface more resistant to fatigue cracking.

Reduction of the feedwater-to-reactor water temperature differential may require both system redesign and operational changes to eliminate unheated feedwater in the sparger-nozzle region during low power and other operations when the main feedwater heaters are not in service.

Since water passing through the feedwater nozzle can originate from sources other than the feedwater system, these sources (such as the reactor water cleanup system) may also need to be examined. One particular source of cold water worthy of mention is the previously mentioned high pressure coolant injection system. An acceptable method of eliminating this source of cold water has not been established.

Because of the importance of this issue, the nuclear industry will need to devote considerable attention to investigate and implement system and operational changes to reduce feedwater to reactor water temperature differentials during all modes of operation. For example, if the feedwater is always maintained within 150°F to 250°F of reactor water temperature, the thermal stresses due to temperature fluctuation within the vessel and feedwater temperature transients will be reduced, the number of fatigue cycles to crack initiation will be significantly increased, and the thermal component of startup-shutdown stresses that could cause cracks to grow will be significantly reduced. Such an increased effort will also need to investigate potential system changes to eliminate the need for frequent sparger and thermal sleeve removals for inspection and repair of feedwater nozzles.

One proposal that has been considered as a long term solution is the welding of the thermal sleeve to the feedwater nozzle to preclude bypass leakage of cold water between the thermal sleeve and the nozzle. At this time the NRC staff has reservations regarding the efficacy of this solution. Weld cracking could result from thermal and/or vibratory stresses in spite of analytical and design efforts to minimize them. The main concern with the welded design, however, is that neither the sleeve-to-nozzle weld nor the nozzle bore are accessible for liquid penetrant examination. It is recommended that alternative designs with better inspectability be considered. For example, two utility groups are installing baffles to restrict water circulation in the nozzle blend radius region.

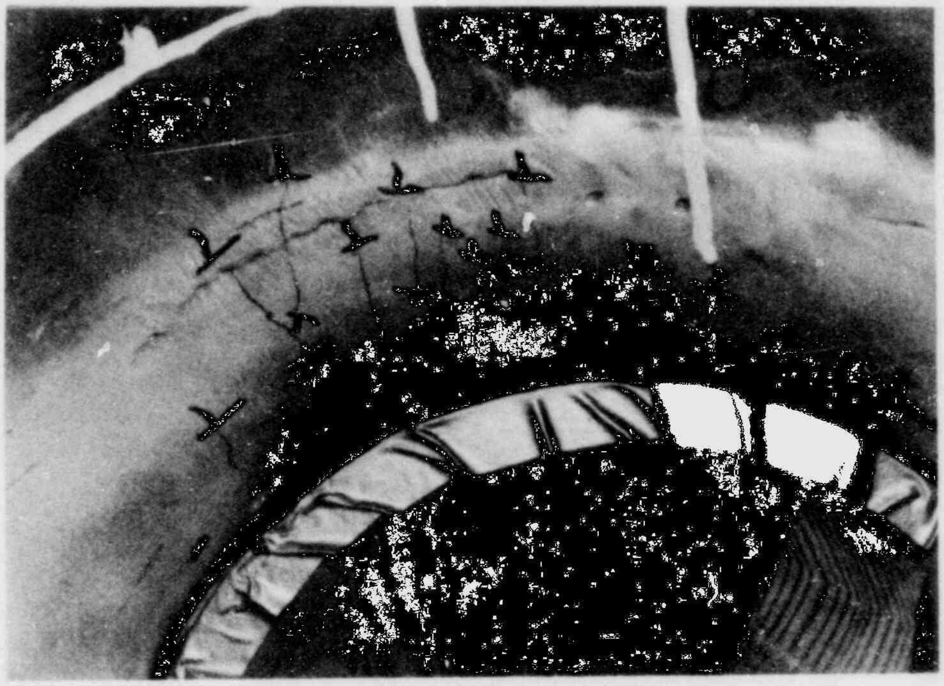
Bypass flow should be limited by a tight fit of the thermal sleeve in the nozzle. Other thermal sleeve designs incorporating piston rings or bolted flanges to eliminate bypass leakage should be considered.

Another approach towards the minimization of this problem concerns the removal of all cladding in the feedwater nozzle blend radius and bore regions. Analyses submitted to the NRC indicate that clad removal can significantly increase nozzle fatigue life. In this approach, the end result would be a nozzle with a clean, smooth surface without flaws or damage from the clad removal process. Several utilities with operating reactors have already decided to implement clad removal.

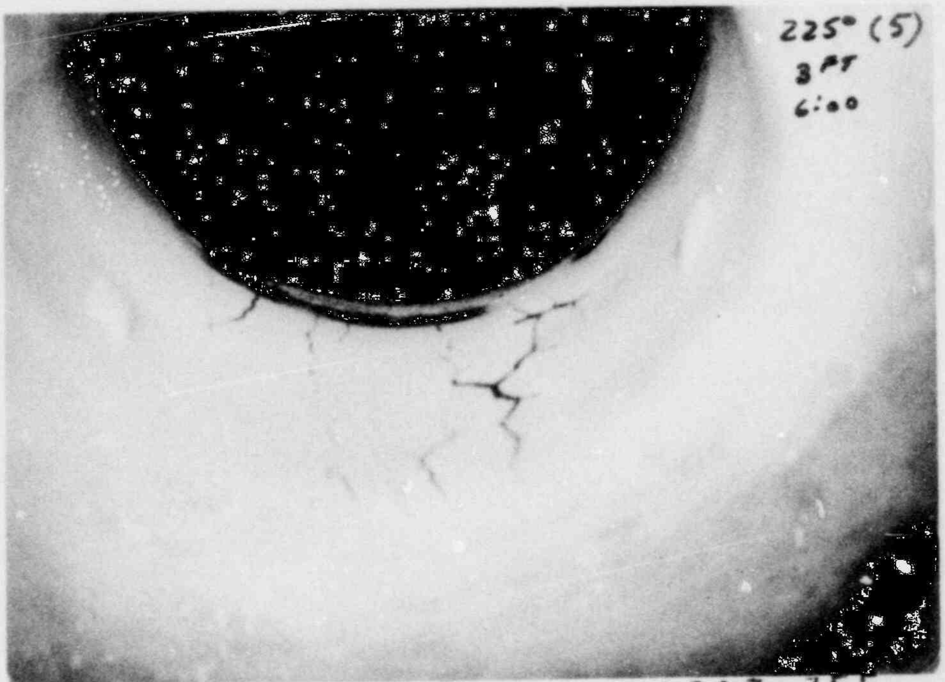
The question of introducing a corrosion problem as a result of removing considerable stainless steel cladding was considered by both GE and the staff. It was concluded that there will be no problem; carbon steel has been used in contact with reactor water with no adverse effects. New BWR reactor vessel nozzles will not be clad. Initially, cladding was applied to minimize rust accumulation in the reactor water thereby maintaining visibility during refueling and minimizing radioactive corrosion product carry-over into the clean-up system. The relatively small area of exposed carbon steel will not impact the above objectives significantly.

In summary, because a large number of operating BWRs either have been or are likely to be found with feedwater nozzle cracks that require repair in the near future, we recommend increased effort to develop improved repair procedures that leave a more fatigue-resistant nozzle surface condition. The present short term procedure is to grind out cracks as they are found. This leaves the nozzle with an irregular surface, sometimes with local areas of abusively ground clad, and base metal exposed to reactor water.

While the procedure has been acceptable as an interim measure, the staff believes that the continuation of this approach for the long term is inappropriate.



Attachment 1.a

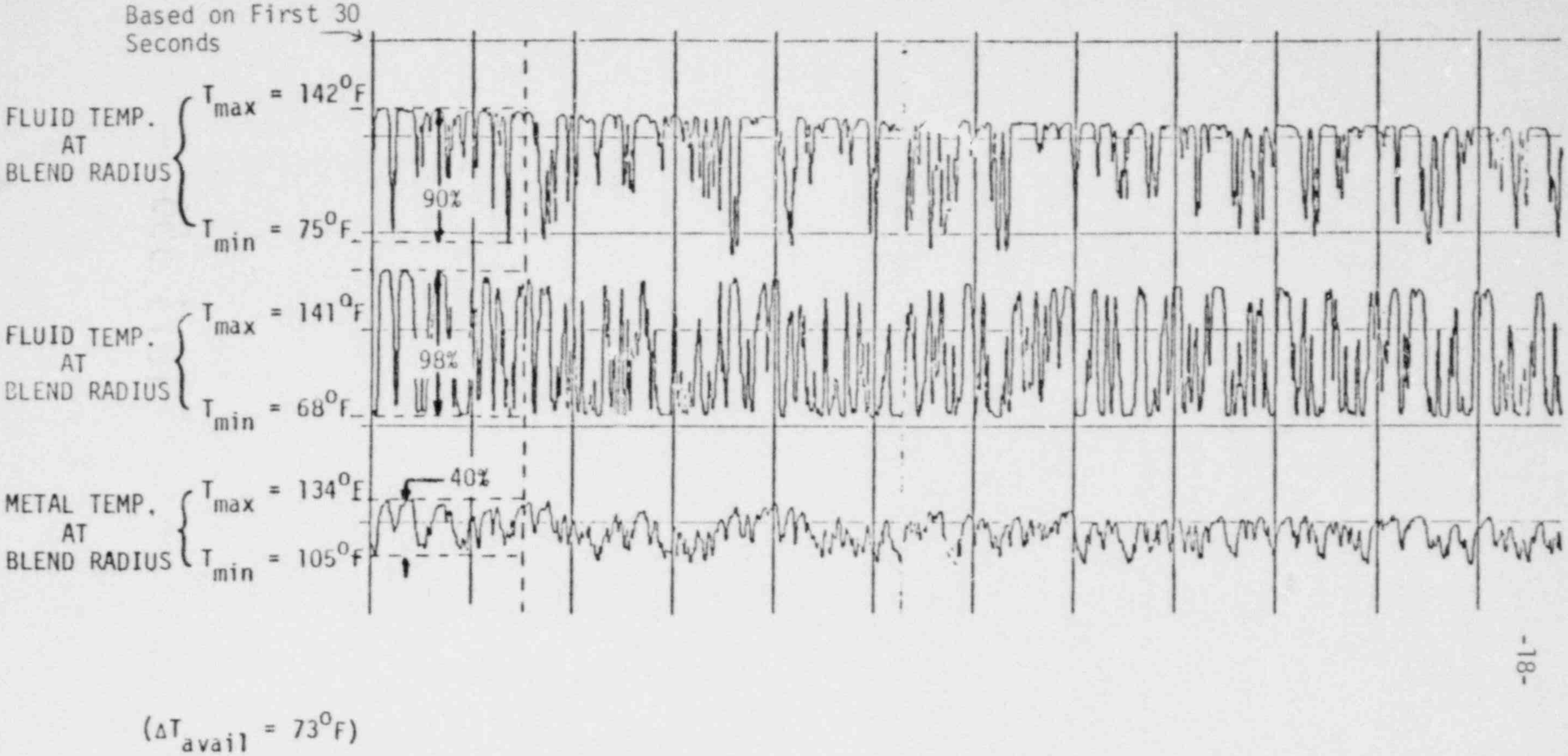


1813 355

Attachment 1.b

<u>Attachment 1.a</u> - Typical example of light cracks on a feedwater nozzle.

<u>Attachment 1.b</u> - Heavy feedwater nozzle cracking in a plant different from Attachment 1.a.



TIME HISTORY TRACES FOR HIGH LEAKAGE/LOW SPARGER FLOW CASE

ATTACHMENT 2

1813 356

SUMMARY OF BWR FEEDWATER NOZZLE CRACKING PROBLEMS*

PLANT	FIRST OPN.	START UPS	FEEDWATER NOZZLE INSPECT.	ACTION TAKEN	GREATEST TOTAL CRACK DEPTH
Dresden 1	10/59	no nozzles - plant design radically different.			
Big Rock Point	10/62	not relevant - reactor vessel has different design			
Humboldt Bay	4/63	110	1976 (TV), '77	Install new sparger Remachine nozzle	3/4 in.
LaCrosse	7/67	no nozzles - feedwater enters recirc. line pump intake			
Nine Mile Pt. 1	11/69	109	1976 (UT), '77	Remachine nozzles (4) Install 4 new spargers	1-1/2in.
Oyster Creek	9/69	97	1976 (UT), '77	Same as Nine Mile	1/2 in.
Dresden 2	4/70	125	1975, '76	Grind out cracks Replace spargers	1/2 in.
Millstone 1	11/70	134	1974, '75, '76	Grind out cracks and replace spargers	0.55 in.
Dresden 3	7/71	93	1975	(same)	3/8 in.
Monticello	3/71	91	1975	(same)	1/2 in.
Quad Cities 1	4/72	112	1976	(same)	0.4 in.
Browns Ferry 1	10/73	68	1975	Grind out cracks, repair spargers	5/32 in.
Browns Ferry 2	8/75	36	1975	Repair spargers; no nozzle cracks	1/32 in.
Quad Cities 2	5/72	102	1975	Grind out cracks and replace spargers	3/8 in.
Vermont Yankee	9/72	61	1975	(same)	0.35 in.
Peach Bottom 2	2/74	65	1976, '77	(same)	3/8 in.
Peach Bottom 3	9/74	46	1977	Grind out cracks	0.04 in.

* Not including nine foreign BWR plants, at least two of which reported cracking

PLANT	FIRST OPN.	START UPS	FEEDWATER NOZZLE INSPECT.	ACTION TAKEN	GREATEST TOTAL CRACK DEPTH
Fitzpatrick 1	2/75	50	1977 (UT), '78	Plan to remachine nozzles, install new spargers	
Cooper	5/74	55	1976	Grind out cracks and replace spargers	0.175 in.
Pilgrim	7/72	69	1976	(same)	3/4 in.
Browns Ferry 3	9/76	21	Insp. planned at first outage (late 1978)		
Hatch 1	11/74	85	1977	Grind out cracks	0.04 in.
Brunswick 2	4/75	62	No f/w nozzles inspection to date		
Duane Arnold	5/74	57	1977 (UT)	None	0
Brunswick 1	10/76		No f/w nozzle inspection to date		

PART II

CONTROL ROD DRIVE RETURN LINE NOZZLE CRACKING

1.0 INTRODUCTION

There is one control rod drive (CRD) return line nozzle in BWR reactor vessels, generally located from 68 inches to 100 inches above the top of the active fuel. The return line is typically 4 inches in diameter. As early as 1974 a General Electric task force on austenitic stainless steel piping noted the large measured thermal gradient in CRD return line (CRD RL) nozzles. Based on the unexpectedly high top to bottom thermal gradients in the nozzle, particularly at low flows, crack initiation susceptability was cited and rerouting the return line was considered. In addition, recent experience with BWR feedwater nozzles has demonstrated the occurrence of crack initiation in nozzles from thermal cycling and further suggested the need to examine CRD return line nozzles. GE issued Service Information Letter (SIL) No. 200 in October 1976 recommending inspection of the nozzle and rerouting of the return line. This SIL was amended in March 1977 to provide for valving out the return line as an interim fix.

The staff has maintained an active involvement in this area through meetings with the General Electric Company and in case-by-case reviews of CRD RL nozzle inspection results from a number of operating BWRs. The following discussion and interim criteria have been developed from currently available information and are subject to future modification. Such interim criteria are needed and have been used to justify continued operation of boiling water reactors.

2.0 STATEMENT OF THE PROBLEM

Dye penetrant (PT) inspections of the CRD return line nozzles to date at domestic BWR plants have revealed cracks in three of the four plants

1813 360

IMAGE EVALUATION
TEST TARGET (MT-3)

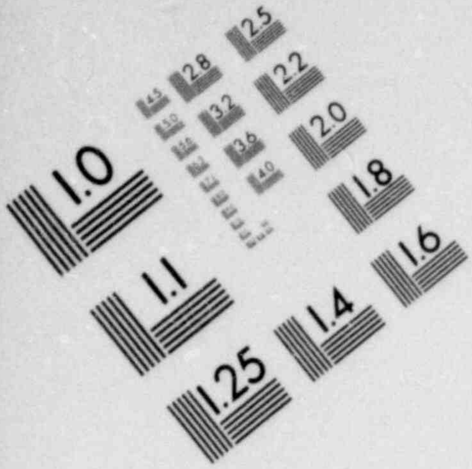
6″

MICROCOPY RESOLUTION TEST CHART

IMAGE EVALUATION
TEST TARGET (MT-3)

6″

MICROCOPY RESOLUTION TEST CHART



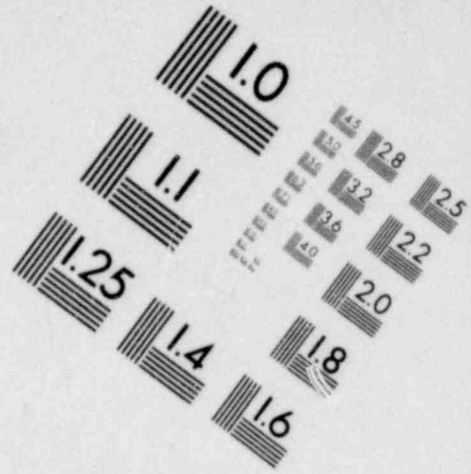
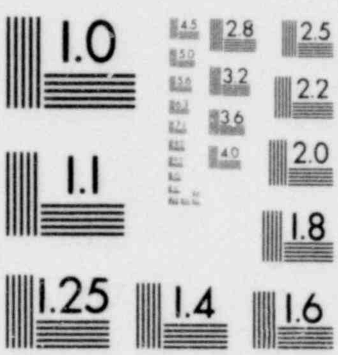
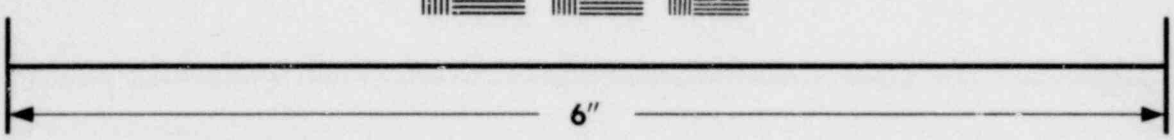
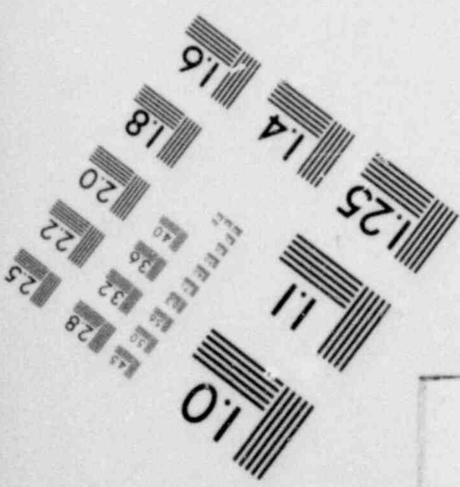


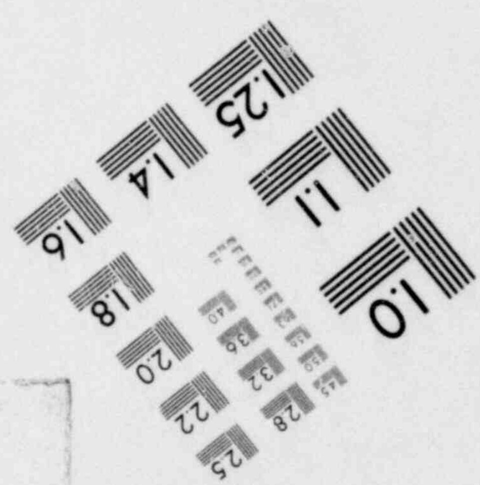
IMAGE EVALUATION
TEST TARGET (MT-3)





MICROCOPY RESOLUTION TEST CHART





inspected. Similar results were found at two overseas reactors. In addition, cracks were found in the reactor vessel wall at Peach Bottom 2 in an area slightly below the CRD return line nozzle but still affected by the return line flow.

The CRD RL nozzle examination results to date are summarized in Table 1. Cracking has been observed in both the blend radius and bore regions of the CRD RL nozzle. While most plants have a thermal sleeve in the CRD RL nozzle, which would be expected to reduce the amount or extent of cracking, cracks have been found at plants with and without sleeves. The cracking observed in the Peach Bottom 2 reactor vessel wall consisted of two horizontal cracks five and seven inches in length and 5 or 6 smaller cracks, located in an area six to twelve inches below the CRD RL nozzle.

3.0 CAUSE OF THE PROBLEM

The underlying cause of crack initiation appears to be thermal fatigue, similar to that experienced with BWR feedwater nozzles. The thermal cycling results from the low temperature (50°F to 100°F) condensate water which enters the reactor vessel through the CRD return line nozzle during normal operation. Although crack initiation mechanisms for the feedwater and CRD RL nozzles appear to be the same, there is a substantial difference in the steady state stresses which ultimately affect crack growth rates.

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CRD RL nozzle crack growth appears to be enhanced by the existence of a continuous large thermal gradient from the top to the bottom of the nozzle, (550°F at the top, 50°F at the bottom), yielding high thermal stresses. It has also been stated, on the basis of stress calculations performed by GE, that control rod scram (which increases CRD return flow from 15 GPM to 60 GPM for about three seconds) and scram testing does not significantly contribute to the nozzle stress distribution.

4.0 REMEDY

Effective long term solutions to this problem require that the thermal cycling in the CRD return line nozzle be eliminated. Accordingly, the General Electric Company has made recommendations, both interim and final, involving system modifications to accomplish this goal.

The interim fix involved (a) valving off the CRD return line to the reactor vessel, (b) reducing CRD RL system flow, (c) raising the exhaust water pressure to a level sufficient to permit the return water to enter the reactor vessel via leakage past the sealing rings in the control rod drives rather than via the return line; (d) adding exhaust water filters and (e) testing of the modified system to verify that the control rod drives would operate properly.

The final system modification proposed rerouting the CRD return line in conjunction with the repair and capping of the nozzle. For BWR/2 plants, GE recommended that the return line be rerouted to the feedwater system outside the primary containment and downstream of all motor operated isolation valves. For BWR/3, 4 and 5 plants the return line could be directed to the reactor water cleanup system downstream of the last motor operated isolation valve.

The reduction of cyclic thermal stresses in the nozzle could also be achieved, it appears, with an effective thermal sleeve. The available evidence from the Nine Mile Point 1 CRD RL nozzle inspection is encouraging in that thorough inspection (after cutting and removing the welded thermal sleeve) revealed no crack-like indications. Since the plant has operated for a significant period of time, being the third domestic BWR to go into operation, the favorable results might indicate that a well-designed thermal sleeve could be an alternative to system modifications, although there are contravening considerations such as the need to periodically inspect the nozzle. At this time, however, it is premature to conclude whether or not such an approach is appropriate when considering our overall safety objectives.

5.0 INTERIM STAFF CRITERIA

Based on the information which is currently available, the NRC staff has determined that the following actions on the part of BWR licensees are appropriate in order to provide a sound basis for continued plant operation:

(1) The CRD return line nozzle and the reactor vessel wall below the nozzle should be inspected at the next scheduled refueling outage by dye penetrant examination, and in general, any crack indications should be repaired, generally, by grinding. The thermal sleeve,

if any, should be removed to permit adequate inspection during this procedure.

(2) Rerouting of the CRD return line to either the RWCU system or feedwater line should be considered by all affected licensees and in those plants where cracking is observed, accomplished at the earliest practical time. The staff recognizes that obtaining the necessary hardware may require long lead times and therefore implementation of the reroute may not be possible at the most immediate upcoming refueling outage. Coincident with rerouting of the return line, the CRD RL nozzle extension should be cut off at the safe end and the nozzle capped. Thermal sleeves should be removed and all cracks removed by grinding. Complete clad removal from the nozzle blend radius and adjacent bore region should be considered, and any weld-build-up areas used for attachment of thermal sleeves should blended smoothly with the nozzle contour.

(3) Implementation of the interim modification proposed by GE, i.e., valving out the return line, should be evaluated by each licensee on a plant specific basis. The incidence of cracking in the CRD RL nozzle, the time necessary to implement rerouting, and the availability of the CRD RL system as a source of makeup water to the reactor vessel should be appropriately considered by each licensee.

(4) The staff should be kept informed on a timely basis of all pertinent activities associated with nozzle inspection and repair and system modifications. Documentation of these activities should be provided promptly for staff review.

6.0 FINAL STAFF POSITIONS

The above interim staff positions are primarily meant to establish a basis for continued plant operation during the time required to develop and implement long term solutions to this problem. As further information becomes available the staff will continue its review in this area and will issue final positions at the appropriate time.

TABLE 1

CRD RETURN NOZZLE

EXAMINATION RESULTS

	YEARS OPN.	START UPS	MAXIMUM CRACK DEPTH (CLAD & BASE)	EXTENT	THERMAL SLEEVE
Peach Bottom 3	2	45	7/8"	General	None
Peach Bottom 2	3	65	*	General; also on vessel wall below CRD RL Nozzle	None
GE Overseas Reactor	6	49	7/8"	General	None
Another Overseas Reactor	~4	~32	9/16"	General	None
Hatch 1	2	85	5/8"	Single Bottom of Nozzle	Expanded Without Flange
Nine Mile Point 1	,	109	--	None	Welded, projects into vessel several inches

*Final results not available; 0.9 inch grind-out to date.

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